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BOUCHEY,G.D. Washington Public Power Supply System
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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards response to draft SER open issues see RP15 & outstanding open branch meeting issues.

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	NRR/DE/HGEB	30	2	2	NRR/DE/MEB	18	1	1
	NRR/DE/MTEB	17	1	1	NRR/DE/QAB	21	1	1
	NRR/DE/SAB	24	1	1	NRR/DE/SEB	25	1	1
	NRR/DHFS/HFEB	40	1	1	NRR/DHFS/LQB	32	1	1
	NRR/DHFS/OLB	34	1	1	NRR/DHFS/PTRB	20	1	1
	NRR/DSI/AEB	26	1	1	NRR/DSI/ASB	27	1	1
	NRR/DSI/CPB	10	1	1	NRR/DSI/CSB	09	1	1
	NRR/DSI/ETSB	12	1	1	NRR/DSI/ICSB	16	1	1
	NRR/DSI/PSB	19	1	1	NRR/DSI/RAB	22	1	1
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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

January 14, 1982
G02-82-41
SS-L-02-CDT-82-018

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SUBMITTAL OF SER OPEN ISSUES

Enclosed are sixty (60) copies of our submittal responding to draft SER open issues and outstanding open branch meeting issues. For ease of review, the pertinent draft SER pages or branch question precedes each issue. A tabulation identifying each item and indicating its resolution status or schedule for close out is also provided.

Very truly yours,

GD Bouchey

G. D. Bouchey, Deputy Director
Safety & Security

CDT/ct
Enclosures

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

B001
S1/60

8201260266

Mr. P. L. Tedesco
Page Two

3. This item is answered by both above items 1 and 2.

Very truly yours,

Original Signed By:

G. D. Bouchey
Director, Nuclear Safety

bk

cc: N. S. Reynolds, D&L
W. Woods, NUS
M. C. Lynch, US NRC
B. J. Youngblooc, US NRC

FILE COPY

AUTHOR			FOR SIGNATURE OF:		
SECTION					
FOR APPROVAL OF					
APPROVED					
DATE					

WP 114

Open SER Issue

3.9.5.c <u>NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking</u>

Open SER Issue

3.9.5.c <u>NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking</u>

A response to this issue was submitted January 13, 1982, in letter GO2-82-36.

of structural integrity or impairment of function. The design procedures and criteria used by the applicant in the design of the WNP-2 reactor internals comply with Standard Review Plan Section 3.9.5 and constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4, and 10.

3.9.6 Inservice Testing of Pumps and Valves

In Sections 3.9.2 and 3.9.3 of this Safety Evaluation Report we discussed the design of safety-related pumps and valves in the WNP-2 facility. The design of these pumps and valves is intended to demonstrate that they will be capable of performing their safety function (open, close, start, etc.) at any time during the plant life. However, to provide added assurance of the reliability of these components, the applicants will periodically test all its safety-related pumps and valves. These tests are performed in general accordance with the rules of Section XI of the ASME Code. These tests verify that these pumps and valves operate successfully when called upon. Additionally, periodic measurements are made of various parameters and compared to baseline measurements in order to detect longterm degradation of the pump or valve performance. Our review under Standard Review Plan Section 3.9.6 covers the applicant's program for preservice and inservice testing of pumps and valves. We give particular attention to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the revised rules of 10 CFR, Part 50, Section 50.55a, paragraph (g). The applicant has not yet submitted its program for the preservice and inservice testing of pumps and valves; therefore, we have not yet completed our review. Any requests for relief from ASME Section XI should be submitted as soon as possible.

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure

Open SER Issue

3.9.6 Preservice and Inservice Testing Programs

A response to this issue was submitted October 1, 1981, by letter number GO2-81-322.

GC Sorensen-440 Chrono File
JC Martin-927M kf/file
BA Holmberg-904A CDT/LB
RG Matlock-901A GDB/LB
WC Bibb-901A sf 2
RM Nelson-906D
CD Taylor-906D
GD Bouchey-396

WPPSS CORRESPONDENCE NO. ____________

October 1, 1981
G02-322

Docket No. 50-397

Mr. A. Schwencer
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2
PUMP AND VALVE TEST PROGRAM PLAN

Enclosed are sixty (60) copies of the Pump and Valve Test Program Plan for WNP-2. These changes will be incorporated into an amendment within four months.

Very truly yours,

G. D. Bouchey
Director, Nuclear Safety

FILE COPY

Enclosure

GDB/CDT/ldm

cc: WS Chin - BPA
AD Toth - NRC Resident
NS Reynolds - Debevoise & Liberman
JC Plunkett - NUS Corporation
R Auluck - NRC DC
OK Earle - B&R RO
EF Beckett - NPI
WNP-2 Files

AUTHOR:	C. D. Taylor		FOR SIGNATURE OF:	G. D. Bouchey
SECTION				
FOR APPROVAL OF	RM Nelson	GC Sorensen	BA Holmberg	RG Matlock
APPROVED	RGJ for RMN	GC Sorensen	BA Holmberg	RG M
DATE	9/22/81	9/24/81	9/29	9/29

DSER 3.9.6 (MEB-49)

on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems, thus causing the inter-systems LOCA.

Pressure isolation valves are required to be Category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes

QUESTION NO. 49

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specification which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

RESPONSE

The valves which separate the Reactor Coolant System (RCS) from interfacing low pressure systems are listed in Table I.

These valves are included in the WNP-2 Pump and Valve Inservice Testing Program which was developed in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV. The Supply System's position is that the requirements of the Code provide adequate assurance of valve integrity. Specifically:

A) The Supply System will leak rate test the valves listed in Table I at least every two years (IWV-3422). This position is justified by the following:

   1. All the valves listed in Table I have direct monitoring position indication which verifies valve position in the Control Room.

   2. The low pressure portions of these interfacing systems are protected against an intersystem LOCA by the following:

      a) The normal functional differential pressure forces the check valves on their seats. The air operator of these testable check valves cannot open the valves at normal differential pressure. (HPCS-V-5, LPCS-V-6, RHR-V-41A, B, C, RHR-V-50A, B, RCIC-V-66).

      b) Electrical interlocks prevent the motor-operated valves from opening when the differential pressure across the valve exceeds specified limits (LPCS-V-5, RHR-V-42A, B, C) or when the RCS pressure exceeds specific values (RHR-V-53A, B, RHR-V-8, RHR-V-9, RHR-V-23, RHR-V-123A, B).

      c) Whenever excessive leakage is present at a pressure boundary isolation valve, this leakage will increase pressure in the downstream side of these systems which will annunciate a high pressure alarm.

      d) Excessive leakage will be channeled into the suppression pool where an increase in suppression pool level will be indicated.

      e) The high pressure core spray pump suction piping is protected by an additional check valve on the pump discharge.

B) The Supply System will specify the leak test medium and the test acceptance criteria as permitted by the ASME Code (IWV-3425 & 3426).

C) The periodic leak test will be done prior to entering Operational Condition 2.

D) After maintenance which is deemed by the Owner to affect leak tightness of the valve, leak testing will be performed in accordance with ASME Section XI prior to the valve's returning to service.

The above positions are consistent with LaSalle County Station's positions which have been approved in LSCS's SER.

Summation - The NRC needs to have their IST people meet with the Supply System. NRC will set up a meeting and get back to us.

A supplemental response was provided on January 8, 1982 (G02-82-15).

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WNP-2 Files

THIS LETTER SATISFIES COMMITMENT NO. ________

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO. ________

January 8, 1982
G02-82-15
SS-L-02-PLP-82-001

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch no. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
ISSUE #49 FROM THE MECHANICAL ENGINEERING
BRANCH/WNP-2 SER MEETING SEPTEMBER 29-30, 1981

As requested by Mr. A. Capucci of your staff, seven (7) copies and supporting drawings describing the WNP-2 response to the subject issue are forwarded.

Should you have any questions regarding this response, please contact Mr. R.M. Nelson, Project Licensing Manager, WNP-2.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

PLP/jca
Enclosure

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

AUTHOR:	PL Powell 1/6/82		FOR SIGNATURE OF:	GD Bouchey
SECTION				
FOR APPROVAL OF	RM Nelson	BA Holmberg	GC Sorensen	
APPROVED		DC Timmins for		
DATE		1/6/82	1/6/82	

measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

We will report the resolution of these issues in a supplement to the Safety Evaluation Report.

CONTAINMENT SYSTEMS BRANCH

ISSUE 6

NRC: Commit to a leakage limit on suppression pool suction line valves.

Supply System: The Supply System will provide a leakage limit criteria of 1 gpm per valve.

Resolution: Marked up FSAR section 6.2-6.3 and Table 6.2-16 attached.

designed such that the penetration is subject to the ILRT test conditions and leakage from these penetrations will be included in the overall leakage rate measured during the ILRT. Testing of the personnel access lock is described in detail in 3.8.2.7.5.

The combined leakage rate of all penetrations and valves subjected to Type B and Type C shall not exceed 60% of the maximum allowable leakage rate, La determined during the ILRT at the calculated peak containment pressure 34.7 psig.

6.2.6.3 Containment Isolation Valve Leakage Rate Test (Type C Tests)

Containment isolation valve leakage rate tests shall be performed by local pressurization in accordance with the requirements of 10CFR50, Appendix J and ANSI N 45.4-1972. The pressure shall be applied in the same direction as that when the valve would be required to perform its safety function unless it can be determined that the results from the test for a pressure applied in a different direction will provide equivalent or more conservative results. The test methods identified in 6.2.6.2 may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustment. Type C isolation valve leakage testing shall normally be performed at 34.7 psig except for the MSIV's which will be leak checked at a pressure between (20 and 30 psig) as defined in 5.4.5.

Table 6.2-16 lists all containment isolation valves on process pipelines penetrating primary containment. Type C testing of these valves shall be considered acceptable provided that the combined leakage rate for all penetrations and valves subject to Type B and C tests is less than 60% of the maximum allowable leakage rate, La, determined during the ILRT at the calculated peak containment pressure, Pa, 34.7 psig. Except for the main steam isolation valves, VALVES SEALED WITH A FLUID FROM A SEALED SYSTEM, and those isolation valves identified in 6.2.3 as potential bypass leakage paths, the isolation valves are not required to meet individual leakage rate limits but as a group must meet the acceptance criteria stated above.

The main steam isolation valves are assigned a specific maximum leakage rate which is within the capability of the MSIV leakage control system (see 6.7).

THE ISOLATION VALVES, SEALED WITH FLUID FROM A SYSTEM WITH A FLUID INVENTORY SUFFICIENT TO ASSURE THE SEALING FUNCTION FOR A MINIMUM OF 30 DAYS AT A PRESSURE OF 1.10 Pa, ARE ASSIGNED A MAXIMUM LEAKAGE RATE OF 10 GPM. THESE VALVES ARE IDENTIFIED IN TABLE 6.2-16.

The isolation valves identified as potential bypass leakage paths will be tested to ASME Code, Section XI, Table IWV standards. Potential bypass leakage (see 6.2.3.3) following a design basis LOCA is significantly less than the value allowed by 15.6.5.

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
HPCS to Reactor	6	3.2-7 6.2-31L	55	A	HPCS-V-5	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					HPCS-V-4	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	17	9						
LPCS to Reactor	8	3.2-7 6.2-31L	55	A	LPCS-V-6	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					LPCS-V-5	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	27	22						
HPCS pump suction from suppression pool	31	3.2-7 6.2-31n	56	B	HPCS-V-15	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	18	18	3	Yes	W	Valves	R.B.	No	18 24
X LPCS pump suction	34	3.2-7 6.2-31n	56	B	LPCS-V-1	MO Gate	O	AC	AC	46	Manual	O	O	O/C	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	~~18~~ 48 24
HPCS test line	49	3.2-7 6.2-31f	56	B	HPCS-V-23	MO Globe	O	AC	AC	F,A	RM	C	C	C	AS-IS	12	Std	6	Yes	W	Valves	R.B.	No	18
HPCS pump min. flow					HPCS-V-12	MO Gate	O	AC	AC	38	RM	C	C	O/C	AS-IS	4	4	53						
HPCS suction relief					HPCS-RV-14	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	65						19
HPCS discharge relief					HPCS-RV-35	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	70						19
LPCS test line	53	3.2-7 6.2-31f	56	B	LPCS-V-12	MO Globe	O	AC	AC	F,V	RM	C	C	C	AS-IS	12	Std	4	Yes	W	Valves	R.B.	No	18
LPCS pump min. flow					LPCS-V-11	MO Globe	O	AC	AC	38	RM	O	C	O/C	AS-IS	3	Std	87						
LPCS suction relief					LPCS-RV-31	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	25						19
LPCS discharge relief					LPCS-RV-18	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	50						19
SLC to Reactor	13	3.2-5 6.2-31m	55	A	SLC-V-7	Check	I	Process	Process	-	-	C	C	C	-	1-1/2	-	-	No	W	Valves	R.B.	No	
					SLC-V-6	Check	O	Process	Process	-	-	C	C	C	-	1-1/2	-	6						
					SLC-V-4A	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21
					SLC-V-4B	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	No.
DW Service Line	92	9.2-4 6.2-31L	56	B	DW-V-157	Gate	I	Manual	Manual	-	-	LC	LC	LC	-	2	-		No	W	Valves	S.B.	.13	
					DW-V-156	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	5						
RHR Condensing Mode Steam Supply	21	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	Yes	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-64	MO Gate	O	DC	DC	X	RM	C	C	C	AS-IS	10	16	2						
RCIC Turbine Steam Supply	45	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	No	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-8	MO Gate	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4	Std	2						
RCIC Pump Minimum Flow	65	3.2-8 6.2-31h	56	B	RCIC-V-19	MO Globe	O	DC	DC	33	RM	C	C	O/C	AS-IS	2	5	7	No	W	Valves	R.B.	No	22
RCIC Turbine Exhaust	4	3.2-8 6.2-31n	56	B	RCIC-V-68	MO Gate	O	DC	DC	35	Manual	O	O	O/C	AS-IS	10	Std	10	No	S	Valves	R.B.	No	[illegible]
RCIC Turbine Exhaust Vacuum Breaker	116	3.2-8 6.2-31l	56	B	RCIC-V-110	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	9	No	A	Valves	R.B.	No	17
					RCIC-V-113	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	5						
RCIC Vacuum Pump Discharge	64	3.2-8 6.2-31q	56	B	RCIC-V-69	MO Gate	O	DC	DC	36	Manual	O	O	O/C	AS-IS	1-1/2	Std	4	No	W	Valves	R.B.	No	22
RCIC Pump Suction from Suppression Pool	33	3.2-8 6.2-31n	56	B	RCIC-V-31	MO Gate	O	DC	DC	32	Manual	C	C	O/C	AS-IS	8	Std	2	No	W	Valves	R.B.	No	23 48
RPV Head Spray	2	3.2-8 6.2-31e	55	A	RCIC-V-66	Check	I	Process	Process	-	-	C	O	O/C	-	6	-	-	No	W	Valves	R.B.	No	3
					RCIC-V-13	MO Gate	O	DC	DC	34	RM	C	O/C	O/C	AS-IS	6	15	2	No	W	Valves	R.B.	No	
					RHR-V-23	MO Globe	O	DC	DC	L,U, M,R	RM	C	O/C	C	AS-IS	6	Std	7	Yes	W	Valves	R.B.	No	

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RHR Loop A:	47	3.2-6 6.2-31p	56	B																				
pump test line					RHR-Y-24A	MO Globe	O	AC	AC	F,Y	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25A	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	33	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55A	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	22	Yes	S	Valves	R.B.	No	18, 19
heat exch. condensate					RHR-V-11A	MO Gate	O	AC	AC	F,Y	RM	C	O/C	C	AS-IS	4	-	18	Yes	W	Valves	R.B.	No	18
heat exch. condensate relief					RHR-RV-36	Relief	O	PP	Spring	-	-	C	C	C	-	8	-	20	Yes	W	Valves	R.B.	No	18, 20
pump minimum flow					RHR-FCV-64A	MO Globe	O	AC	AC	38	RM	C	C	O/C	AS-IS	3	15	22	Yes	W	Valves	R.B.	No	[illegible], 18
heat exch. thermal relief					RHR-RV-1A	Relief	O	PP	Spring	-	-	C	C	C	-	1-1/2	-	188	Yes	W	Valves	R.B.	No	18, 19
heat exch. vent					RHR-V-73A	MO Globe	O	AC	AC	39	Manual	C	O/C	C	AS-IS	2	Std	175	Yes	A	Valves	R.B.	No	18
FDR system inter-tie					RHR-V-121	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	3	-	6	No	W	Valves	R.B.	No	
CAC system Loop A drain					RHR-V-134A	MO Gate	O	AC	AC	37	Manual	C	C	O/C	AS-IS	.2	Std	44	Yes	W	Valves	R.B.	No	18
pump A suction relief					RHR-RV-88A	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18
RHR Loop B	48	3.2-6 6.2-31p	56	B																				
pump test line					RHR-V-24B	MO Globe	O	AC	AC	F,Y	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25B	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55B	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	20	Yes	S	Valves	R.B.	No	18, 19
pump A&B suction relief					RHR-RV-5	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	20	Yes	W	Valves	R.B.	No	18, 19

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	COC	Code Co. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RHR Loop A Suppression Pool Suction	35	3.2-6 6.2-31n	56	B	RHR-V-4A	MO Gate	O	AC	AC	46	Manual	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	48
RHR Loop B Suppression Pool Suction	32	3.2-6 6.2-31n	56	B	RHR-V-4B	MO Gate	O	AC	AC	46	Manual	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	48
RHR Loop C Suppression Pool Suction	35	3.2-6 6.2-31n	56	B	RHR-V-4C	MO Gate	O	AC	AC	46	Manual	O	O	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	48
RHR Loop A:	117	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RY-95A	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	24	Yes	S	Valves	R.B.	No	18, 19
condensate pot drain					RHR-V-124A	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	11	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-124B	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	12	Yes	W	Valves	R.B.	No	18
RHR Loop B:	118	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RY-95B	Relief	O	PP	Spring	-	-	C	C	C	-	10	-	21	Yes 19	S	Valves	R.B.	No	18
condensate pot drain					RHR-V-125A	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	17	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-125B	MO Gate	O	AC	AC	39	Manual	C	C	C	AS-IS	1-1/2	Std	14	Yes	W	Valves	R.B.	No	18
RHR Loop C:	26	3.2-6 6.2-31f	56	B																				
pump test line					RHR-V-21	MO Globe	O	AC	AC	F,Y	RM	C	C	C	AS-IS	18	Std	34	Yes	W	Valves	R.B.	No	18
discharge header relief					RHR-RY-25C	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18, 19
pump C suction relief					RHR-RY-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	38	RM	O	C	O/C	AS-IS	3	15	30	Yes	W	Valves	R.B.	No	18
Suppression Pool Spray Loop A	25A	3.2-6 6.2-31h	56	B	RHR-V-27A	MO Gate	O	AC	AC	F,Y	RM	C	C	O/C	AS-IS	6	Std	5	Yes	W	Valves	R.B.	No	2, 18, 24
Suppression Pool Spray Loop B	25B	3.2-6 6.2-31h	56	B	RHR-V-27B	MO Gate	O	AC	AC	F,Y	RM	C	C	O/C	AS-IS	6	Std	6	Yes	W	Valves	R.B.	No	2, 18, 24

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Cl. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RCC Inlet Header	5	3.2-14 6.2-31t	56	B	RCC-V-104	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	5	No	W	Valves	R.B.	No	17
					RCC-V-5	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3						
RCC Outlet Header	46	3.2-14 6.2-31o	56	B	RCC-V-21	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3	No	W	Valves	R.B.	No	
					RCC-V-40	MO Gate	I	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	-						
Suppression Pool Cleanup Suction	100	3.2-12 6.2-31l	56	B	FPC-V-153	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	2	No	W	Valves	R.B.	No	17,48 48
					FPC-V-154	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	7						
Suppression Pool Cleanup Return	101	3.2-12 6.2-31o	56	B	FPC-V-156	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	3	No	W	Valves	R.B.	No	17,48 48
					FPC-V-149	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	6	-	41						
RWCU From Reactor	14	3.2-11 6.2-31k	55	A	RWCU-V-1	MO Gate	I	AC	AC	A,J, E,W	RM	O	O	C	AS-IS	6	Std	-	No	W	Valves	Rad. W.	.35	
					RWCU-V-4	MO Gate	O	DC	DC	A,J, E,Y,W	RM	O	O	C	AS-IS	6	Std	4						
RRC Pump A seal Water	43A	3.2-3 6.2-31c	56	B	RRC-V-13A	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16A	MO Gate	O	AC	AC	45	Manual	O	O	O	AS-IS	3/4	Std	2						
RRC Pump B seal water	43B	3.2-3 6.2-31c	56	B	RRC-V-13B	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16B	MO Gate	O	AC	AC	45	Manual	O	O	O	AS-IS	3/4	Std	2						
RRC Sample Line	77Aa	3.2-3 6.2-31d	55	A	RRC-V-19	SO Globe	I	AC	Spring	A,C	RM	C	C	C/O	C	3/4	<5	-	No	W	Valves	T.B.	.05	
					RRC-V-20	AO Globe	O	Air	Spring	A,C	RM	C	C	C/O	C	3/4	Std							

TABLE 6.2-16 (Continued)

40. Normally closed. Signalled to open if reactor building pressure exceeds wetwell pressure by 0.5 psid. Valves automatically reshut when the above condition no longer exists. Operator to use valve position indicator as confirmation of valve status.

41. Indication of containment air compressor discharge header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-20 should the containment air compressors become unavailable. The isolation check valve, CIA-V-21, provides immediate isolation.

42. Indication of nitrogen bottle header pressure and a low pressure alarm exist in the main control room. The operator can remote-manually shut valve CIA-V-30(A,B) should the nitrogen bottle bank pressure decrease below the alarm setpoint. The isolation check valves, CIA-V-31(A,B) provide immediate isolation.

43. The operator's indication that remote-manual closure of the TIP shear valves is required, is failure of the TIP ball valves to close as monitored on Panel S.

44. Normally closed. Opened only when testing wetwell to drywell vacuum breakers.

45. The isolation valve can be remote-manually closed upon indication that the CRD or the RRC pumps have been tripped. The isolation check valves, RRC-V-13 (A, B), provide immediate isolation.

46. These valves are the ECCS and drywell spray suction and discharge isolation valves. ECCS operation is essential during the LOCA period; therefore, there are no automatic isolation signals. The valve closure requirement will be indicated by a high level alarm in the appropriate reactor building sump, which will be indicative of excessive ECCS leakage into secondary containment.

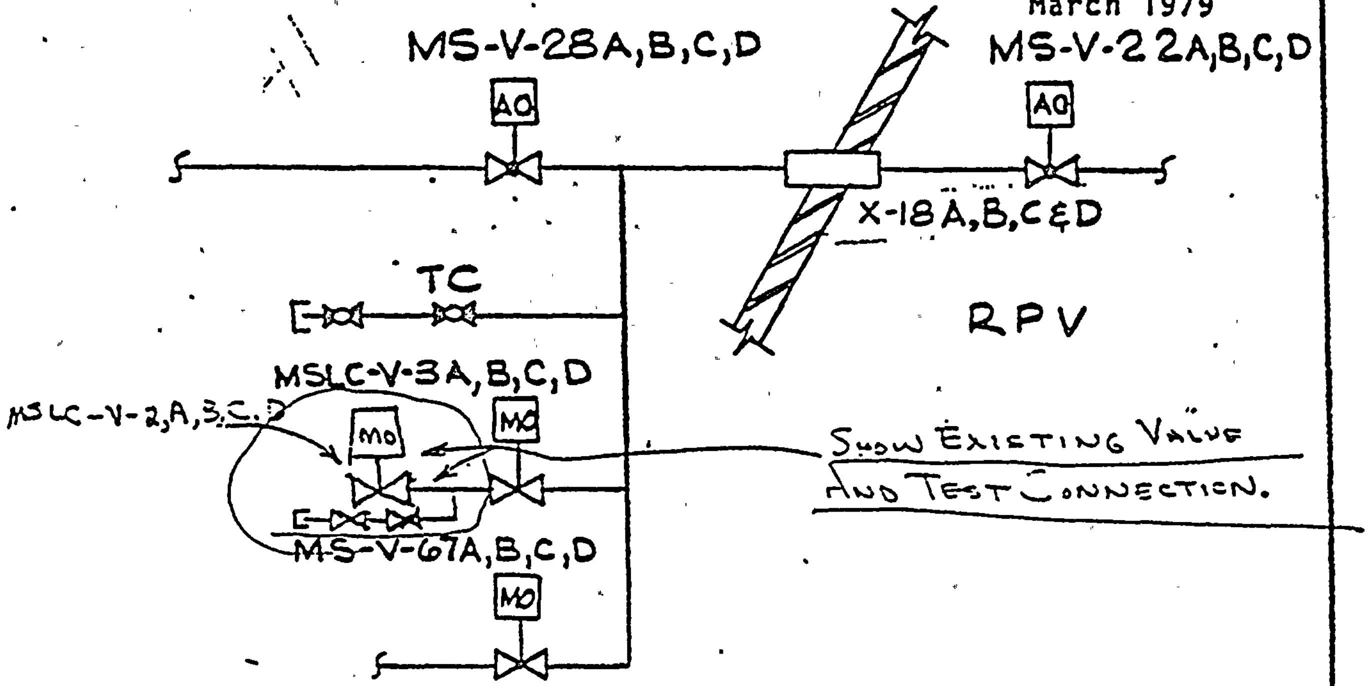
47. The isolation valve can be remote-manually closed upon indication that the RWCU pumps have been tripped. The reactor feedwater isolation check valves provide immediate isolation.

48. THE ISOLATION VALVE WILL BE TESTED WITH WATER. THE MAXIMUM ALLOWABLE LEAKAGE RATE IS 1.0 GPM PER VALVE WHICH WILL NOT BE INCLUDED IN THE COMBINED TYPE B+C LEAKAGE RATE.

CONTAINMENT SYSTEMS BRANCH

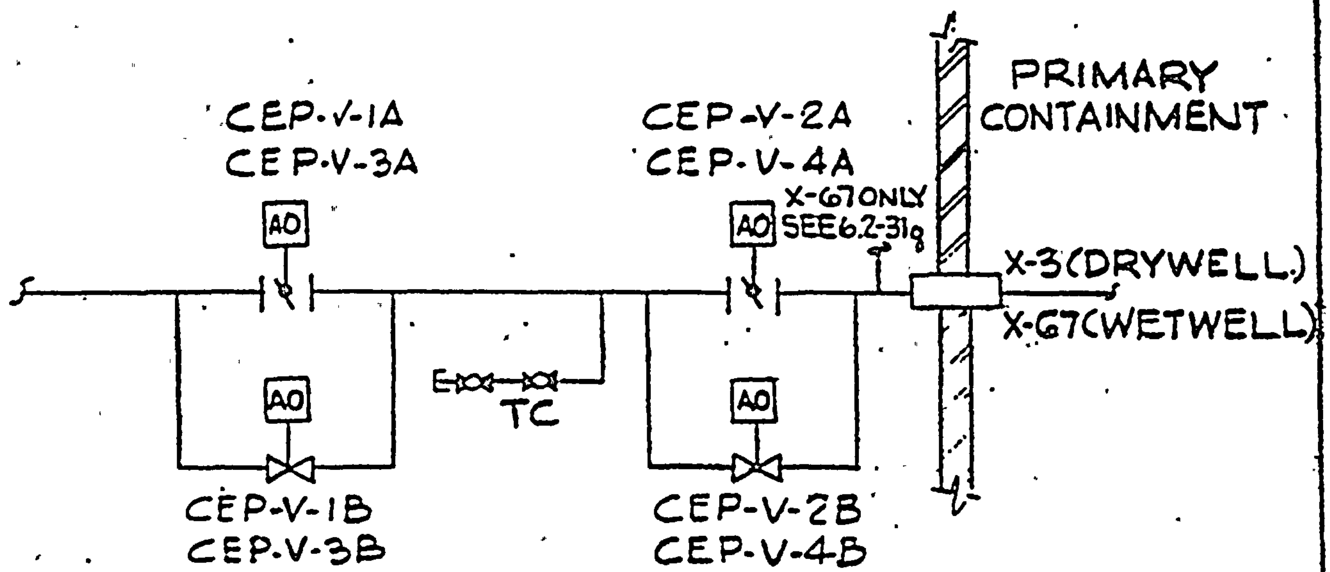
ISSUE 7

NRC:	Figure 6.2-31j - are all four LCS valves on mainsteam isolation valves tested at the same time?
Supply System:	The Supply System will revise Figure 6.2-31j to show a capability to test the valves individually.
Resolution:	Revised Figure 6.2-31j attached.



NOTE: SEE NOTES 3 AND 8 ON FIG 6.2-31a

MAIN STEAM LINES



NOTE: SEE NOTES 4 AND 8 ON FIG 6.2-31a

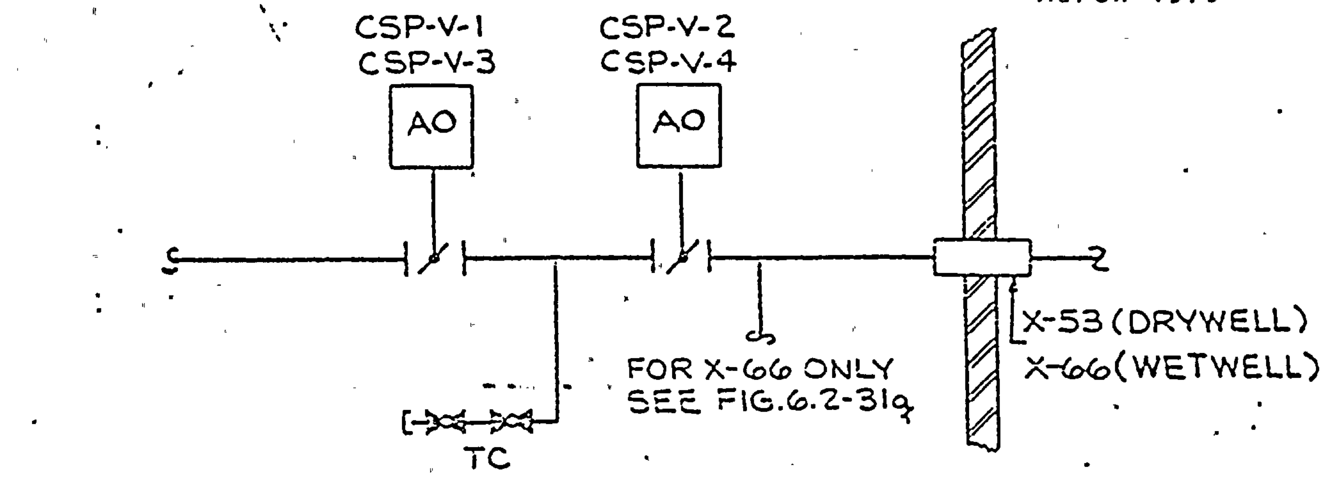
X-3 DRYWELL PURGE EXHAUST
X-67 WETWELL PURGE EXHAUST

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-18A, X-18B, X-18C, X-18D, X-3 & X-67	FIGURE 6.2-31j

CONTAINMENT SYSTEMS BRANCH

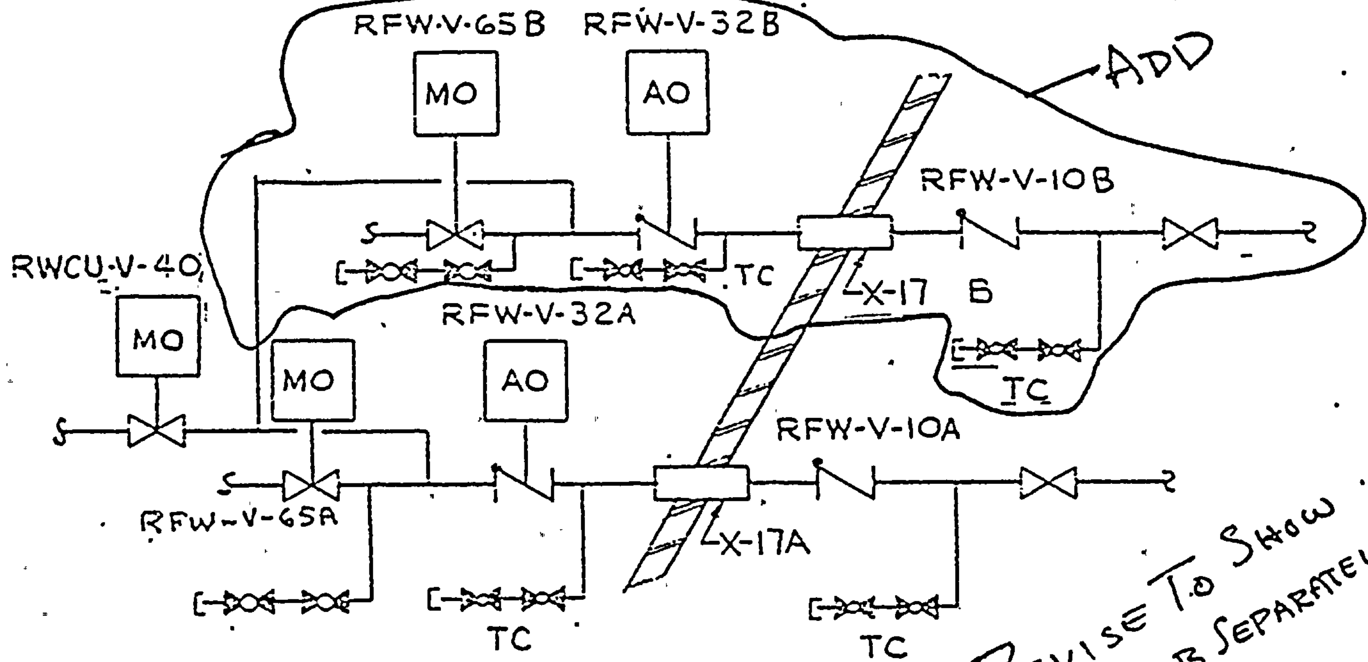
ISSUE 8

NRC:	Figure 6.2-31b, feedwater line RWCU-V-40 How is this valve to be tested?
Supply System:	The Supply System will revise Figure 6.2-31b to indicate test connections available for RWCU-V-40.
Resolution:	Revised Figure 6.2-31b attached.



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

REVISE TO SHOW
PENET. 17A + 17B SEPARATELY
AND SHOW X-TIE

REACTOR FEEDWATER LINES

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

DSER 3.10

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

Our evaluation of the adequacy of the applicant's program for qualification of electrical and mechanical equipment important to safety for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT), which consists of reviewers from the Equipment Qualification Branch (EQB) and consultants from Idaho National Engineering Laboratory (INEL), has reviewed the methodology and procedure of equipment seismic and dynamic qualification program contained in the pertinent FSAR Sections 3.7, 3.9.2, 3.9.3, 3.10 and Appendices 3.10A, B, C. The SQRT has concluded that, with one important exception, the information contained in the FSAR does meet the intent of the current licensing criteria as described in IEEE 344-1975, Regulatory Guides 1.92 and 1.100, and the Standard Review Plan Sections 3.9.2 and 3.10. The exception being that the effect of hydrodynamic vibratory loads (associated with either safety relief valve discharge or LOCA blowdown into the suppression pool), on equipment liable to experience this kind of excitation, is not addressed in the applicant's FSAR. The applicant is required to consider the hydrodynamic loading effect on equipment susceptible to this kind of loading in the seismic and dynamic qualification program. Furthermore, IEEE Std 344-1975 covers the seismic qualification aspects; however, the aging and test sequence aspects of the equipment qualification must be in accordance with the requirements of IEEE Std 323-1974.

In our communication with the applicant, we indicated that a substantial portion (85%-90%) of the equipment must be qualified, documented in an auditable manner, and installed onsite before an onsite audit by the staff can be performed. We also indicated to the applicant the type of information necessary for us to select the equipment items for a detailed onsite review. Once the applicant has indicated that his work is substantially complete the staff will conduct an onsite audit shortly thereafter. We shall report the results of our audit in a future supplement to our SER. Our review of this area will be complete after the applicant has demonstrated the adequacy of his qualification program through a satisfactorily audit.

Open SER Issue

3.10 Hydrodynamic Vibratory Loading Effect

The response to this issue will be included in the Equipment Qualification Program Submittal.

2. operation beyond Cycle 1 is not permitted until stability analysis is provided and approved for the additional cycles of operation;
3. the natural circulation operating mode is not permitted; and
4. the core flow should be checked at least once per day and the average power range monitor flow biased scram calibrated at least once per month to account for possible effects of crud deposition.

The above restrictions should be incorporated into the proposed Technical Specifications, except for Item 2 which should be incorporated as a license condition.

In addition, the following open item should be resolved prior to issuance of the operating license:

- the operating limit MCPR calculated by including the ODYN methods must be provided for review and approval.

We conclude that, with the exceptions noted above, the thermal-hydraulic design of the core conforms to the requirements of General Design Criterion 10 of 10 CFR Part 50, Regulatory Guides 1.68 and 1.133, and Section 4.4 of the Standard Review Plan and is, therefore, acceptable.

4.4 MCPR Operating Limit Calculation by ODYN

The operating limit minimum critical power ratio (OLMCPR) has been established using ODYN methods for rapid pressurization transients. The Supply System responses to Reactor Systems Branch questions 211.049 and 211.084 committed to the ODYN analyses. FSAR Section 4.4 (Table 4.4-1), 5.2.2, 15.0, 15.1.2, 15.2.2 and 15.2.3 have been modified to reflect the ODYN analyses. (Refer to SER item 15. ODYN Reanalysis)

DSER 4.4

1. A description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

2. A description of how the operators will be trained in the purpose and implementation of the system.

We will review the applicants conformance evaluation report when it becomes available, consistent with our plans for review of the operating plants. Any action resulting from our review will be applied at that time. On this basis, we find the LPMS acceptable for an operating license.

Summary

The staff has reviewed the thermal-hydraulic design of the core as described in Section 4.4 of the FSAR for WNP-2. The scope of the review included the design criteria, implementation of the design criteria as presented by the final core design, and the steady-state analysis of the core thermal-hydraulic performance. The applicant's thermal-hydraulic analyses were performed using approved methods and correlations and found acceptable. However, the operating license should be restricted to the following conditions:

1. single loop operation is not permitted unless supporting analyses are provided and approved;

SER Open Item - Thermal Hydraulic Section - Core Performance Branch
Section 4.4 Thermal Hydraulic Design Evaluation

NRC POSITION

Reference: Memo L. S. Rubenstein, Assistant Director for Core and Plant Systems, DSI to R. L. Tedesco, Assistant Director for Licensing, DL, SER Input for Thermal and Hydraulic Design of the Core for WNP-2 Power Plant, December 1, 1981.

Prior to release of the SER, the applicant should provide a written commitment to evaluate the Loose Parts Monitoring System (LPMS) in accordance with the Regulatory Guide 1.133, Revision 1 (May 1981) on a schedule specified by the applicant. The conformance evaluation report should emphasize the programmatic aspects such as establishing the alert level, the operator training in the purpose and implementation of the LPMS, and diagnostic procedures used to confirm the presence of a loose part.

WNP-2 POSITION

Letter G02-81-335, G. D. Bouchey to Schwencer, Loose Parts Detection System Conformance Report, October 5, 1981 confirms that the WNP-2 Loose Parts Detection System has been designed and specified to meet the requirements of Regulatory Guide 1.133 Revision 1, May 1981. In addition, the Supply System will prepare a conformance evaluation report which emphasizes the programmatic aspects of implementing the detection program.

The report will discuss the program that the Supply System will be implementing. This program will include:

1) A description of the operator training program for the system. This program will include discussion on the purpose and function of the LPMS.

2) A description of the operating and diagnostic procedures used when the system is operated in the manual mode - listening to audio signals from all installed sensors. Alert levels will be established.

3) A description of the operating and diagnostic procedures used when the system is operated in the automatic mode. Alert levels will be established.

4) A description of the Technical Specifications for operation of the system.

5) The reporting requirements necessary if a loose part is confirmed.

The conformance report will be submitted for review prior to the time of initial reactor startup testing.

DSER 4.6

Based on our evaluation, we conclude that these systems, taken together, satisfy the requirements of General Design Criteria 20, "Protection System Functions," 26, "Reactivity Limits," and 28,"Reactivity Control System Redundancy and Capability," as noted above.

The CRDS is capable of providing reactivity control following postulated accidents with an appropriate margin for a stuck rod. This capability is demonstrated by the loss-of-coolant accident and rod dropout analyses presented by the applicant which, in turn, show that the consequences are acceptable and core cooling is maintained, as required by General Design Criteria 20, "Protection System Functions," 27, "Combined Reactivity Control Systems Capability," and 28, "Reactivity Limits."

The instrumented volume (IV) is closely coupled hydraulically to the scram discharge volume (SDV) by sloping the SDV toward the IV and increasing the pipe sizes over previous SDV section's pipe sizes. At the connection of the SDV and IV the SDV is an eight-inch pipe and the IV is a 12-inch pipe. Therefore filling of the SDV without an indication in the IV is not possible, such as by a slow or partial loss of air pressure to the scram discharge valves. Any water leaking into the SDV would enter the IV and exit through the drain. Leakage in excess of the drain's capacity would fill the IV and result first in an alarm followed by a rod block and then a reactor scram if the problem was not corrected in time.

[The applicant has not yet responded to our request for additional information in a letter dated May 5, 1981 regarding our Office of Analyses and Evaluation of Operational Data (AEOD) report entitled, "Safety Concerns Associated with a Pipe Break in the BWR Scram System." The report describes a poten-

tial sequence of events which could result from a break in the BWR scram discharge piping during a scram condition concurrent with an inability to reclose the scram outlet valves. We will report resolution of this item in a supplement to this SER.]

Open SER Issue

4.6 Safety Concerns Associated with Pipe Break

A response to this issue was submitted January 13, 1982, by letter G02-82-37.

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

January 13, 1982
G02-82-37
SS-L-02-CDT-82-017

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO REQUEST FOR INFORMATION

Enclosed are sixty (60) copies of the Supply System 120-day response to the NRC's concern regarding Pipe Breaks in the BWR Scram Discharge Volume (see Reference 1 to the Attachment).

Also, this is the Supply System's response to Generic Letter 81-34 (see Reference 4 to the Attachment).

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

DSER 4.6

[The applicant has not responded to our concern relating to the effects on the safety and operability of the control rod drive hydraulic system if the drive or cooling water control valves fail closed or fail open. We will report resolution of this item in a supplement to this SER].

The applicant has responded to the concerns identified in the NRC generic study, "BWR Scram Discharge System Safety Evaluation" dated December 1, 1980. The applicant identified the need to upgrade the scram discharge control system in four areas.

1. Addition of redundant vent and drain isolation valves
2. Addition of redundant and diverse level instrumentation for scram
3. Relocation and piping of instrument piping directly to the scram instrument volume
4. Addition of new surveillance and operating procedures.

We have reviewed the applicant's responses and find the applicant has demonstrated compliance with the required upgrading to the generic safety evaluation. Based on our review, we conclude that the scram discharge system meets the requirements of the NRC generic study, "BWR Scram Discharge System Safety Evaluation" dated December 1, 1980 and is, therefore, acceptable, pending confirmatory receipt of an acceptable revised Section 4.6 of the FSAR to reflect these modifications and confirmation by the Office of Inspection and Enforcement that the four aforementioned modifications have been installed.

[Based on our review, we cannot conclude that the control rod drive system is not acceptable regarding the control valve failure or scram system pipe break until the applicant provides the information discussed above. We will provide our evaluation of the resolution of these items in a supplement to this SER].

Open SER Issue

4.6 CRD Hydraulic System (Q. 10.43)

A response to Q. 10.43 was submitted on December 18, 1981, in letter G02-81-533.

Q. 010.043
(4.6)

Describe the effects on the safety and operability of the control rod drive hydraulic system if the following control rod drive system valves either fail closed or fail open:

1) Drive water pressure control valve (between F060 and F061);

2) Cooling water pressure control valve (between F070 and F071).

Response:

The function of the F003 pressure control valve (PCV) is to provide a means of adjusting the drive water header and cooling water header pressures. The F003 PCV is a manually controlled motor-operated valve which is controllable from the main control room. Indicating lights are provided in the control room for the valve full open and full closed positions. Adjustment of the F003 PCV in concert with adjustments to the F002 flow control valve permit adjustment of the drive water header pressure to approximately 260 psi above vessel pressure while at the same time, maintaining the drive cooling water header pressure at approximately 20 psi above vessel pressure.

If the F003 PCV were to fail to a full open position, the cooling water pressure would increase and the drive water pressure would decrease. The resulting cooling water pressure increase could cause control rods to drift inward. The existence of rod drifts would be alarmed to the control room operator for appropriate action. The resulting drop in drive water pressure would make normal control rod notch movements impossible but would not affect the ability of the scram function.

Conversely, if the F003 PCV were to fail to a full closed position, the cooling water pressure would decrease while the drive water pressure would increase. The reduction in cooling water pressure (and flow) would eventually lead to high CRD temperatures being alarmed in the control room. The CRD system's scram function would not be affected by the increase in drive water pressure. In the limiting case, the resulting increase in drive water pressure would reach up to the shut off pressure of the supply pump (1750 psig). The occurrence of this condition during withdrawal of a drive at zero reactor

pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations and tests indicate that the drive would accelerate from 3 inches per second to no more than 6.5 inches per second. The rod movement would stop after the driving signal is removed or rod block is enforced by the Reactor Manual Control System (RMCS). In the unlikely event where RMCS fails to enforce a rod block, the peak fuel enthalpy for drive speeds of 6.5 inches per second is well below the fuel cladding failure threshold design limit. Therefore, due to provisions in the system design and margin in the fuel design, this postulated scenario will not compromise the integrity of the fuel.

In both of the cases described above, the manually operated bypass PCV (F004) in conjunction with the isolation gate valves located upstream and downstream of the F003 PCV would enable the operators to take corrective action.

In conclusion, although the failure to the full open or full closed position of the drive/cooling water PCV will cause perturbation in the CRD system operation, it does not present a safety problem or affect the scram capability of the CRD system.

The PCV F005 was deleted from the CRD hydraulic system in the process of implementing the CRD return line deletion modifications, therefore, this question is not applicable.

See response to Question 211.130 for additional details on the deletion of the CRD return line.

DSER 5.2.2

WNP-2 OPEN ITEMS

1. Overpressurization Protection (5.2.2)- The applicant must submit for our reivew and approval, a plant specific overpressurization analysis using the ODYN code and including the effect of recirculation pump trip.

2. Safety/Relief Valve Surveillance (5.2.2)- The applicant must commit to participate in a surveillance program to monitor the performance of safety/relief valves.

3. Pressure Interlocks on ECC Injection Valves (6.3)- The applicant must verify that interlocks are present at all times for both manual and automatic valve operation and that the interlocks do not allow valve opening until the reactor coolant pressure is below the design pressure of the ECC system involved, or provide an alternative configuration which satifies the requirements of SRP Section 6.3.

4. Premature LPCI Diversion (6.3)- The applicant must provide assurance that LPCI flow will not be diverted to containment cooling before adequate core cooling is provided. (We have accepted a discussion of emergency procedures and operator training for this item on other applications.)

5. Long Term Air Supply to ADS Valves (6.3)- The applicant must verify that the bottled air supply serving as a backup to the normal air supply for the ADS valves is valved in during normal operation, or provide justification as to why credit should be given to this air supply.

6. Thermal Power Monitor in Transient Analyses (15)- We require that the thermal power monitor time constant be included in the plant technical specifications or that no credit be taken for the thermal power monitor in transient analyses.

5.2.2 Overpressurization Protection (RSB-1)

Refer to ODYN reanalysis (SER Section 15, RSB-7) for a response to this issue.

5.2.2 Safety/Relief Valve (RSB-2)

Refer to LRG submittal response to RSB-28. (Appendix I)

3 CONCLUSIONS

Our technical evaluation has not identified any practical methods by which the existing WNP-2 reactor vessel can comply with the specific requirements of Paragraphs III.B.1, III.B.3, III.B.4, III.C.1, III.C.2, IV.A.2.a, IV.A.3, and IV.B of Appendix G and Paragraph II.B of Appendix H, 10 CFR Part 50. Alternate methods justify an exemption for Paragraphs III.B.1, III.B.3, and III.B.4 of Appendix G. Paragraphs III.B.1, III.C.1, III.C.2, IV.A.2.a, IV.A.3 and IV.B of Appendix G and Paragraph II.B of Appendix H will remain open items until the applicant submits data to demonstrate compliance.

Based on the foregoing, pursuant to 10 CFR, Section 50.12, exemptions from the specific requirements of Appendices G and H of 10 CFR Part 50, as discussed above, are authorized by law and can be granted without endangering life or property or the common defense and security and are otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendices G and H of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Furthermore, we have determined that the granting of these exemptions does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that these exemptions would be insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5 (d)(4) that an environmental impact statement, or negative declaration and environments appraisals, need not be granted in connection with this action.

5.3.2 Pressure-Temperature Limits

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as

Open SER Issue

5.3.1 <u>10CFR50 Appendices G and H Compliance</u>

A response to this issue was submitted December 18, 1981, by letter number G02-81-532.

THIS LETTER SATISFIES COMMITMENT NO. ______

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO. ______

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NR Reynolds D&L
WNP-2 Files

December 18, 1981
G02-81-532
SS-L-02-CDT-81-107

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
APPENDIX G AND H INFORMATION RESPONSES TO MATERIALS ENGINEERING BRANCH - COMPONENT INTEGRITY SECTION

Reference: Letter, R. Auluck to R.L. Ferguson, "WNP-2 FSAR - Request for Additional Information", dated September 8, 1981

Enclosed are sixty copies of the information on Appendix G and H as agreed with NRC as proposed by the LRG during a meeting April 1, 1981.

Also, enclosed are sixty copies of the responses to Questions 121.011 through 121.019, which were transmitted to the Supply System via the referenced letter. These questions will be incorporated into an amendment to the WNP-2 FSAR.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

FILE COPY

AUTHOR: CD Taylor			FOR SIGNATURE OF: GD Bouchey	
SECTION				
FOR APPROVAL OF	RM Nelson	BA Holmberg	GC Sorensen	
APPROVED	RM Nelson	BA Holmberg	GC Sorensen	
DATE		12/15	12-17-81	

DSER 6.2.1.8.e (CSB-43)

a description of the specific application of the generic Mark II pool dynamic loads and methods for WPPSS Nuclear Project No. 2 and the plant unique loads used in assessing the capability of the WPPSS Nuclear Project No. 2 containment and components to pool dynamic phenomena.'

A summary of our review status for each of the pool dynamic loads is presented in Table 6.2. This table provides a description of each load or phenomenon, the Mark II Owners Group's load specification, and references our review status and the applicants' position on each load.

As indicated in Table 6.2, the applicants agreed to adopt all but three of our generic criteria. These items relate to steam condensation oscillation and chugging loads (Load I.C.2 in Table 6.2) and quencher air clearing loads (Load II.B in Table 6.2). Alternative criteria were proposed by the applicants for these items. Our evaluation of these alternative criteria is provided below.

6.2.1.8.e <u>Steam Condensation Oscillation Load</u> (Load I.C.2 in Table 6-2)

In its letter no. G02-81-239 dated August 13, 1981, the applicants indicated that the generic condensation oscillation load specification definition developed for the Mark II Owners Group, and acceptable for those plants with reinforced concrete containments, has been determined to be excessive for WPPSS Nuclear Project No. 2 plant.

The applicants contend that, based on the examination and evaluation of available test data, condensation oscillation loading is less critical than the chugging load and does not represent a governing load for structures, piping and equipment in WPPSS Nuclear Project No. 2.

The applicants indicated that a detailed report will be submitted by December 15, 1981 summarizing the results of these studies. We will report on our findings regarding this load specification upon completing our review of the pertinent information.

6.2.1.8.f <u>Steam Condensation Chugging Load</u> (Load I.C.2 in Table 6-2)

In July 1981, a report titled "Chugging Loads - Revised Definition and Application Methodology for Mark II Containments (Based on 4TCO Test Results)," was submitted by the applicants in lieu of the generic chugging load methodology found acceptable by the staff in NUREG-0808. The application methodology for WPPSS Nuclear Project No. 2 containment accounts for the plant-specific parameters governing the response such as vent length, three-dimensional multivent suppression pool geometry with sloped bottom and the flexibility of suppression pool structural boundary. Seven key chugs, having significantly larger pressure peaks and more power than the remaining chugs, obtained from the 4TCO tests were chosen to envelope all 4TCO data base at all frequencies. These key chugs together with thirteen chugging traces from the same time windows to which the key chug occurs are used to deduce seven single vent impulsive acceleration sources used to develop the revised chugging load definition.

Each single vent impulsive acceleration source is applied in-phase at exit elevation of the three vents in each of the thirty-four radial lines where downcomers are located in the WPPSS Nuclear Project No. 2.

6.21.8e

WNP-2

ISSUE 43

NRC:

The NRC will not be able to review the WNP-2 condensation oscillation report, scheduled to be submitted by December 15, 1981, by the SER date. The only way to meet the SER and SSER date, WNP-2 will have to accept the generic condensation oscillation load specification. Any plant unique data may require a NRC review of 6 to 12 months.

Supply System:

Condensation Oscillation Report was submitted to NRC on December 24, 1981, by letter number G02-81-552.

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docket file
kf/file-906D
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BAH/lb
RMN/lb
RGM/lb

GDB/lb
sf 2
pf 1

THIS LETTER SATISFIES COMMITMENT NO. ______

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO: ______

FILE COPY

Docket No. 50-397

December 24, 1981
G02-81-552

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
CONDENSATION OSCILLATION LOAD FOR WNP-2

Reference: G02-81-239, August 31, 1981, G. D. Bouchey, Supply System, to R. L. Tedesco, NRC

The Reference letter advised you that the Mark II Owners group generic condensation oscillation (CO) load definition was excessive for the WNP-2 plant, that preliminary studies indicated CO loading should not be a governing load for plant assessments, and that a final report summarizing the results of these studies would be submitted by the end of 1981. This issue was discussed further with the NRC staff and consultants at the Containment Systems Branch review meeting in Richland during the week of September 14, 1981, and is currently identified as an open item in the draft SER for WNP-2.

Transmitted herewith is the final report, "Comparison of Condensation Oscillation and Chugging Loads for Assessment of WPPSS Nuclear Project No. 2", prepared on our behalf, by Burns and Roe, Inc. This report summarizes results of studies performed to evaluate data from single-vent tests (4TCO), to review results of multi-vent tests (JAERI), and to compare CO and chugging loads. As concluded in the report, these studies confirm that the CO load does not represent a governing load for WNP-2, and consequently need not be considered in assessments of structures, piping, and equipment.

AUTHOR	EA Fredenburg E. Fredenburg		FOR SIGNATURE OF:	GD Bouchey
SECTION				
FOR APPROVAL OF	BA Holmberg	RG Matlock	RM Nelson	
APPROVED	E Fredenburg	[illegible]	[illegible] / for RMN	
DATE	for KAH	"	12/23/81	

Please note that both a proprietary and a nonproprietary version of the CO report are enclosed. The proprietary version includes figures which have been designated proprietary by JAERI, the General Electric Company, and by Burns and Roe, Inc. Consistent with the provisions under which this information was made available to the Supply System and to Burns and Roe, and as attested to by the attached affidavit, we request that the proprietary version of this report be withheld from public disclosure in accordance with the provisions of 10CFR2.790.

We would be pleased to meet with the NRC staff and consultants, at your earliest convenience, to respond to any questions or to further discuss the contents of the enclosed report.

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director,
Safety and Security

EAF:kjf

Enclosure: Summary Report, "Comparison of Condensation Oscillation and Chugging Loads for Assessment of WPPSS Nuclear Project No. 2"

cc:	R. Auluck - NRC	w/proprietary report (1 copy)
	EF Beckett - Nuclear Projects, Inc.	w/o attachment
	WS Chin - BPA	w/o attachment
	AI Cygelman - B&R Site	w/o attachment
	OK Earle - B&R RO	w/o attachment
	F. Eltawila - NRC	w/proprietary report (1 copy)
	R. Feil - NRC Site	w/o attachment
	JA Forrest - B&R RO	w/o attachment
	J. Lehrer - Brookhaven National Lab.	w/proprietary report (2 copies)
	ND Lewis - EFSEC	w/o attachment
	FA MacLean - General Electric	w/o attachment
	JC Plunkett, Jr. - NUS	w/o attachment
	NS Reynolds - D&L	w/o attachment
	S. Smith - General Electric	w/o attachment
	RE Snaith - B&R NY	w/o attachment
	JJ Verderber - B&R NY	w/o attachment
	WNP-2 Files	

DSER 6.2.1.8.g (CSB44,45,46,47a-f)

The chug start times in each radial direction are assigned arbitrarily based on the smallest variance in one thousand Monte Carlo trials drawn from a uniform distribution of start times having a width of 50 milliseconds.

The WPPSS Nuclear Project No. 2 pool pressures thus obtained are compared against JAERI data and found to bound the JAERI data.

The staff and its consultant, Brookhven National Laboratory, has completed its review of the applicants' improved load methodology and found it to be acceptable. The staff will issue a NUREG report to discuss its findings regarding all WPPSS Nuclear Project No. 2 plant-unique loads.

6.2.1.8.g Quencher Air Clearing Load (Load II.B in Table 6-2)

The applicant has committed to install a X-quencher device designed by the General Electric Company. Subsequent to the issuance of NUREG-0487 and in view of the availability of in-plant test data for the X-quencher, the applicants have proposed an alternative to our acceptance criteria set forth in NUREG-0487. The alternative load specification was submitted to the staff in a report titled, "SRV Loads - Improved Definition and Application Methodology for Mark II Containment." The improved load definition was derived from test results obtained from Caorso (Italy) in-plant SRV actuation experiments. Summary of results from Tokai-2 (Japan) in-plant SRV tests were used by the applicants to confirm the adequacy of the improved load definition.

Based on our review of the applicants' improved SRV load definition, we concluded that additional information is required to resolve our concerns. In a meeting that was held with the applicants on September 14-17, 1981, the applicants stated that they will respond to our concerns by December 15, 1981. We will report our findings regarding this item upon receipt of the applicants' response.

In addition to our generic review of the Mark II pool dynamic loads, we have reviewed a limited number of pool dynamic loads on a plant unique basis. The basis of our review of these areas are discussed below.

(1) Drywell Pressure History (Load I.B.1.f first column in Table 6.2)
The drywell pressure history is utilized as part of the overall pool swell load methodology. The applicants have based its calculation of the drywell pressure history on the methods described in General Electric Topical Report NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model," and Appendix B of NEDO-20533. We previously reviewed this methodology on a generic basis and concluded it was acceptable.

(2) Large Structure Impact Loads (Load I.B.3.b in Table 6.2)
The applicant has stated that the WPPSS Nuclear Project No. 2 does not contain any large horizontal structures in the pool swell zone that would be subject to impact loads. Since the applicants have reviewed the as-built plant design and concluded that no large structure exists in the pool swell impact zone, we concur that no load specification is necessary for expansive structures.

(3) Post-Swell Wave Load and Seismic Slosh Load (Load IV.C and D in Table 6.2)
These loads have been determined to be secondary loads in that they are not design controlling. We have reviewed the applicants' evaluations of these loads and find them to be acceptable.

ISSUE 44

NRC: Caorso air volume in discharge line is smaller than WNP-2. Therefore, load is higher. Comparisons between the two are concerns to the NRC.

Supply System: See Issue 47 a-f

ISSUE 45

NRC: Question 022.055. The NRC is concerned about the statement saying maximum pressure for multiple valve actuation is higher than 5.87 psi.

Supply System: See Issue 47 a-f

ISSUE 46

NRC: The NRC believes that DFFR requires the pressure difference (calculated to extrapolate Caorso conditions to WNP-2 conditions) be added to Caorso measured peak pressures, rather than used to calculate a multiplier, as was done in the WNP-2 SRV load definition.

Supply System: See Issue 47 a-f

CONTAINMENT SYSTEMS BRANCH

ISSUE 47a-f Acceptable SRV Design Load Specification

NRC: Acceptance of pressure from Caorso based on 90/90 (9.37 psi) for subsequent actuation with the resolution of the following:

a. Use DFFR correlation to determine a Δ pressure rather than multiplier to account for differences between Caorso test conditions and WNP-2 design conditions, using the WNP-2 design temperature as specified in the Technical Specifications to calculate the Δ pressure.

b. Either account for differences in discharge line air volume between Caorso and WNP-2 or provide justification.

c. Account for the differences in pressure between the multiple and single valve case by adding the difference between the mean pressure from Caorso multiple valve case and the single valve case ~~for~~ the 9.37 psi, or provide justification that the WNP-2 design is conservative.

d. Account for effect of two vacuum breakers vs. one vacuum breaker as used in Caorso by adding the difference between the mean pressure from Caorso test conducted with two vacuum breaker and the tests conducted with one vacuum breaker blocked to the 9.37 psi or provide justification.

e. 1) For multiple valve case at initial actuation, the vertical pressure distribution used as specified in the WNP-2 SRV report is acceptable.

   2) For single valve case, at subsequent actuation, the staff requires the method of NUREG-0487 for vertical pressure distribution or provide justification that the present load design is conservative.

f. Circumferential pressure distribution should be based on NUREG-0487 or as calculated from the Burns & Roe hydrodynamic model.

g. The staff requires in-plant tests (NUREG-0763) to verify ΔT between the bulk and local pool temperature and to verify boundary pressure loads.

Supply System: The Supply System will provide either resolution or justification by December 15, 1981.

Responses to CSB47a-f were submitted January 13, 1982 by letter G02-82-35.

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

January 13, 1982
G02-82-35
SS-L-02-CDT-82-015

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO CSB OPEN ITEMS 44-48

Enclosed are sixty (60) copies of the responses to open items 44 - 48 from the Containment Systems Branch meeting held September 14-17, 1981. These items should be closed by receipt of this submittal.

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

DSER 6.2.1.8.d(CSB-48)
DSER 6.2.1.8.g(5) (CSB-41,47g)

(4) Steam Condensation Submerged Drag Loads (Load III.C. in Table 6.2)
Submerged structures in the WPPSS Nuclear Project No. 2 suppression pool were assessed by the applicant for loads due to main vent steam condensation and due to SRV actuation. A procedure was developed to provide a conservative evaluation of these loads. The approach utilizes the same basic approach that was applied to the generic drag load methodology with several modifications. The source strength for these loads was derived from the 4TCO data. Plant-unique flow fields will be defined consistently with the chugging and condensation oscillation boundary loads developed under items 6.2.1.8e and f above.

For submerged structure drag loads due to SRV actuation, the applicants indicated that data from Caorso SRV tests are examined to define the spatial distribution and to define time history of the dynamic pressure gradient measured across column, vent pipes and SRV discharge line in the Caorso plant.

Based on our review of the applicants' preliminary submittal of these loads, we conclude that it is acceptable pending formal documentation of the information that was presented to us during the September 14-17, 1981 meetings.

(5) Pool Temperature Limit (Phenomenon I.A in Table 6.2) and Safety Relief Valve In-Plant Test
We require in Criterion II.A. of NUREG-0487 that the suppression pool local temperature shall not exceed 210 degrees Fahrenheit for all plant transients involving safety relief valve operations. The applicants have not provided plant-unique analyses for pool temperature responses to transients involving safety relief valve operation. These analyses are currently scheduled for submittal in late December 1981. We will report our findings in a supplement to this draft Safety Evaluation Report.

We have requested the applicants to perform a comprehensive safety relief valve in-plant test which is to be completed prior to commercial operation of the facility. These tests will include single and multiple valve tests to confirm the adequacy of the piping system design. In addition, we have requested the applicants to utilize information from these tests to establish the difference between local and bulk pool temperatures to demonstrate that a maximum local pool temperature specification of 210 degrees Fahrenheit will not be exceeded.

During the September 14-17, 1981 meetings the applicant indicated that they would respond to our request by December 15, 1981. We will report our findings regarding this item in the SER.

In conclusion, we conducted an assessment of the WPPSS Nuclear Project No. 2 against our generic acceptance criteria. We also reviewed those few areas where alternative criteria have been proposed. In addition, we completed our review of pool dynamic loads that were relegated to plant-unique reviews. In each of these areas, we concluded that the pool dynamics load utilized by the applicants are conservative and therefore acceptable, except for:

1 - Steam Condensation oscillation load specification;
2 - Pool temperature transients involving safety relief valve discharge and in-plant SRV test; and
3 - Quencher Air Clearing Load.

CONTAINMENT SYSTEMS BRANCH

ISSUE 48

NRC: The staff believes that the WNP-2 improved chugging load definition is reasonably conservative. The applicant has responded to our concerns in a satisfactory manner.

Preliminary information regarding SRV and LOCA submerged structure drag loads seems to be reasonably conservative.

Supply System: Information provided to the NRC on chugging, ~~and~~ SRV and LOCA submerged structure drag loads was provided for discussion purposes, and will be submitted formally by October 2, 1981.

Response submitted January 13, 1982, by letter G02-82-35.

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

January 13, 1982
G02-82-35
SS-L-02-CDT-82-015

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO CSB OPEN ITEMS 44-48

Enclosed are sixty (60) copies of the responses to open items 44 - 48 from the Containment Systems Branch meeting held September 14-17, 1981. These items should be closed by receipt of this submittal.

Very truly yours,

GD Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

CONTAINMENT SYSTEMS BRANCH

ISSUE 41

NRC: When will NRC receive WNP-2 Pool Temperature Analysis (NUREG-0783).

Supply System: The analysis was submitted on December 15, 1981 by letter G02-81-524.

CONTAINMENT SYSTEMS BRANCH

ISSUE 47 Acceptable SRV Design Load Specification

NRC: Acceptance of pressure from Caorso based on 90/90 (9.37 psi) for subsequent actuation with the resolution of the following:

a. Use DFFR correlation to determine a Δ pressure rather than multiplier to account for differences between Caorso test conditions and WNP-2 design conditions, using the WNP-2 design temperature as specified in the Technical Specifications to calculate the Δ pressure.

b. Either account for differences in discharge line air volume between Caorso and WNP-2 or provide justification.

c. Account for the differences in pressure between the multiple and single valve case by adding the difference between the mean pressure from Caorso multiple valve case and the single valve case ~~for~~ the 9.37 psi, or provide justification that the WNP-2 design is conservative.

d. Account for effect of two vacuum breakers vs. one vacuum breaker as used in Caorso by adding the difference between the mean pressure from Caorso test conducted with two vacuum breaker and the tests conducted with one vacuum breaker blocked to the 9.37 psi or provide justification.

e. 1) For multiple valve case at initial actuation, the vertical pressure distribution used as specified in the WNP-2 SRV report is acceptable.

2) For single valve case, at subsequent actuation, the staff requires the method of NUREG-0487 for vertical pressure distribution or provide justification that the present load design is conservative.

f. Circumferential pressure distribution should be based on NUREG-0487 or as calculated from the Burns & Roe hydrodynamic model.

g. The staff requires in-plant tests (NUREG-0763) to verify ΔT between the bulk and local pool temperature and to verify boundary pressure loads.

Supply System: The Supply System will provide either resolution or justification by December 15, 1981.

Resolution was submitted December 15, 1981, by letter number G02-81-524.

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THIS LETTER SATISFIES COMMITMENT NO. ______

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO. ______

Docket No. 50-397

December 15, 1981
GO2-81-524

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SUPPRESSION POOL TEMPERATURE TRANSIENT
ANALYSIS AND IN-PLANT SRV TEST

The purpose of this letter is to transmit the results of the suppression pool temperature transient analysis for WNP-2, and to advise you that we will perform an in-plant test to measure the difference between local and bulk pool temperatures during main steam relief valve discharge. Both of these were identified as open items during the Containment Systems Branch review meeting in Richland in September 1981, and in the draft SER for WNP-2.

Results of the pool temperature transient analysis are contained in the attached Report No. 14057-U(D)-1, "Suppression Pool Temperature Analysis", prepared for Washington Public Power Supply System Nuclear Project No. 2 by Stone and Webster Engineering Corporation. This analysis indicates that for the cases evaluated, the maximum suppression pool peak bulk temperature is 198^{o}F. For wetwell airspace pressure of 0 psi gage, this is 35^{o}F below saturation temperature at the quencher centerline elevation, and thus allows for a local-to-bulk temperature difference of 15^{o}F, in accordance with the acceptance criteria of draft NUREG 0783 for steam mass flow rates of less than 42 lbs/ft^2/sec.

NUREG 0763 provides guidelines for determining whether plant-specific tests may be required to measure SRV discharge loads and pool temperature gradients. The stated policy is that new plant-specific tests will be required if plant parameters affecting loads and temperature distribution are substantially different from those previously tested. According to NUREG 0763, "applicants may be able to demonstrate that discharge conditions in their plants are sufficiently similar to conditions previously tested to obviate the need for any new tests or to curtail the scope of tests."

AUTHOR: EA Fredenburg E. Fredenburg			FOR SIGNATURE OF: GD Bouchey		
SECTION					
FOR APPROVAL OF	RM Nelson	BA Holmberg	JD Martin	RG Matlock	
APPROVED	[signature]	E. Fredenburg	JD Martin	[signature]	
DATE	12-14-81	for BAH.	12/14/81	12/14/81	

Plant parameters defined in NUREG 0763 which must be evaluated for similarity to parameters previously tested are as follows:

1. discharge device geometry
2. discharge line parameters: line length, area, volume, quencher submergence, vacuum breaker size, available pool area per quencher
3. steam flow rate
4. quencher location and orientation, and pool geometry
5. structural characteristics of containment

In-plant tests performed in the Caorso plant in Italy and in the Tokai plant in Japan are directly applicable to WNP-2. Both Caorso and Tokai have Mark II containments geometrically similar to WNP-2, with SRV discharge line parameters and quencher geometry essentially identical to WNP-2. The bottom of both the Caorso and Tokai containments are flat, while the WNP-2 containment has a trapezoidal-shaped bottom. The Caorso containment is reinforced concrete, while Tokai utilizes a steel containment similar to WNP-2.

Data from both the Caorso and Tokai tests have been extensively evaluated on our behalf by Burns and Roe, Inc. Of the two in-plant tests, Caorso has been the most thoroughly investigated due to the availability of data through the Mark II Owners group.

The quencher air clearing load definition for WNP-2 was based on detailed analysis of data from the in-plant SRV tests in Caorso, and confirmed by evaluation of data from the Tokai tests. Differences between the Caorso plant conditions, and WNP-2, were accounted for in the load definition report submitted to the NRC in August 1980 ("SRV Loads-Improved Definition and Application Methodology for Mark II Containments"). NRC review of the SRV load definition for WNP-2 considered all of the plant parameters identified above which could affect quencher air clearing loads. Differences between Coarso and WNP-2 plant parameters were addressed by the Supply System in response to NRC questions during the licensing process. Following discussions with the NRC during the Containment Systems Branch review meetings in Richland in September 1981, some modifications to the SRV load definition were made to account for minor differences in these parameters.

A summary of comparisons of the Caorso and WNP-2 significant plant parameters potentially affecting the quencher air-clearing load is provided below:

Quencher Geometry

The Caorso and WNP-2 quenchers are essentially identical. (See FSAR question 22.053).

SRV Discharge Line

SRV discharge line lengths, areas, volumes, and quencher submergence for Caorso and WNP-2 are similar. Minor differences are accounted for in the response to CSB issue #47 from the September 1981 review of WNP-2.)

Vacuum Breaker Size

The SRV discharge line vacuum breakers affect the reflood transient within the SRV line, and therefore the internal water level within the line for a subsequent SRV actuation. The Caorso test conditions included a diversity of SRV line initial water levels. Since the WNP-2 SRV load definition envelopes the loads observed at Caorso, differences in vacuum breakers are accounted for. (See FSAR questions 22.054, 22.057, and the response to CSB issue #47 from the September 1981 review of WNP-2.)

Pool Area per Quencher

The pool surface area per quencher for WNP-2 is slightly larger in WNP-2 than in Caorso. This difference would have no significant effect on the quencher air clearing load. (See FSAR question 022.107, and the response to CSB issue #47 from the September 1981 review of WNP-2.)

Steam Flow Rate

The steam flow rates in WNP-2 range from 236 to 252 lbm/sec. Steam flow rates in the Caorso tests ranged from 238 to 244 lbm/sec. The Caorso steam flow rates were within 1% of the flow rates for the six lowest setpoint SRV's at WNP-2, and within 2.5% of the flow rates for the highest setpoint SRV at WNP-2. These differences are not significant.

Quencher Location and Orientation

Quenchers at both Caorso and WNP-2 are arranged around the suppression pool in an inner circle and an outer circle. The outer quenchers in WNP-2 are farther away from the containment wall (9.95 feet) than the outer discharging quencher in the Caorso tests (7.5 feet). Since bubble pressure attenuates with distance, using Caorso test pressures applied directly to the WNP-2 containment is conservative.

Pool Geometry

Except for the trapezoidal-shaped bottom on WNP-2, Caorso and Tokai have essentially identical geometries to WNP-2. Since the magnitude of the SRV quencher air clearing loads acting on the containment wall have been found to be primarily a function of proximity of containment to the quencher, the effect of the shape of the pool bottom is not significant.

Structural Characteristics of Containment

Fluid/structure interaction (FSI) effects during the Caorso tests and the analytical methods used to extract rigid wall pressures from the test measurements are discussed in detail in the SRV load definition report submitted to NRC. As shown therein, the analytical model used to predict boundary pressures in a Mark II containment is in good agreement with Caorso test measurements. Also discussed is the application of FSI effects to the

steel containment structure of WNP-2. Differences between Caorso structural characteristics and WNP-2 are thus accounted for in the SRV load definition. (Also, see FSAR question 22.063.)

Based on this comparison of plant parameters, only minor differences between WNP-2 and Caorso are found to exist, and these differences are conservatively accounted for in the SRV load definition for WNP-2. We have concluded that an in-plant test to confirm the adequacy of the quencher air-clearing load would not substantially add to the body of knowledge already obtained from other in-plant tests, and is therefore not required for WNP-2, per the guidelines of NUREG 0763.

Suppression pool temperature response was also measured in the Caorso tests. As previously mentioned, the only significant difference between WNP-2 and Caorso which could conceivably affect pool temperature gradients is the shape of the pool bottom. Since the local-to-bulk pool temperature difference measured in the Caorso tests, as reported in NEDO-24798, was only 5°F, it does not appear likely that the temperature difference for WNP-2 would approach the allowable value of 15°F determined from the attached report. However, because the NRC has questioned the influence of the trapezoidal-shaped pool bottom on flow characteristics and temperature distribution in the suppression pool during SRV discharge, and because of uncertainties which would be associated with a purely analytical approach to this problem, the Supply System commits to conducting an in-plant test to measure the local-to-bulk temperature difference. Local temperature will be measured by temperature sensors mounted on the containment wall opposite the discharging quencher, in accordance with the guidelines of draft NUREG 0783. The existing suppression pool temperature monitoring system will be utilized in these tests, for measurement of both local and bulk pool temperatures.

G. D. Bouchey
Deputy Director
Safety and Security

EAF:kjf

Enclosure: Report No. 14057-U(D)-1
"Suppression Pool Temperature Analysis"

cc:	R. Auluck - NRC	(w/1 attachment)
	EF Beckett - Nuclear Projects Inc.	(w/o attachment)
	WS Chin - BPA	(w/o attachment)
	AI Cygelman - B&R Site (954W)	(w/o attachment)
	F. Eltawila - NRC	(w/1 attachment)
	R. Feil - Resident Inspector	(w/o attachment)
	JA Forrest - B&R RO	(w/o attachment)
	SA Giusti - BPC (904)	(w/o attachment)
	ND Lewis - EFSEC, Olympia	(w/o attachment)
	FA MacLean - GE, San Jose	(w/o attachment)
	TA Mangelsdorf - BPC (982)	(w/o attachment)
	S. Smith - GE, San Jose	(w/o attachment)
	RE Snaith - B&R NY	(w/o attachment)
	JJ Verderber - B&R NY	(w/o attachment)

that the use of the drywell and suppression pool purge system is acceptable provided that the applicants limit purging to control of containment pressure, inerting and deinerting of the containment and qualifies the valves to the requirements of Branch Technical Position CSB 6-4 as discussed further in Section 22 of the Safety Evaluation Report. It should be noted that the current design does not include a debris screen to protect the valves. The applicants have committed to install the debris screens prior to fuel load.

Conclusion

Based on the above evaluation, we conclude that the applicants' proposed design of the containment isolation system satisfies the requirements of Criteria 54, 55, 56 and 57 of the General Design Criteria and is acceptable except for the recombiner scrubber return line to the suppression pool where we will require the applicant to provide redundant isolation valve to meet GDC 55.

6.2.5 Combustible Gas Control

The combustible gas control systems include piping, valves, and components and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentration of these gases.

The scope of our review of the design and functional capability of the combustible gas control system for WPPSS Nuclear Project No. 2 included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentration, and reduce combustible gas concentrations within the containment following the design basis accident. Our review also included the applicants' proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the system provided to support the adequacy of the design bases.

The bases for our acceptance are the conformance of system design and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards.

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of corrosion of metals in the containment, metal-water reaction between the fuel cladding, and as a result of radiolytic decomposition of the post-accident emergency cooling water. The applicants analyzed the production and accumulation of hydrogen from the above sources in accordance with the guidelines of Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment." The guideline regarding the metal-water reaction states that hydrogen production is five times the maximum calculated reaction under 10 CFR 50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.23 mils, whichever is greater, in two minutes. The applicants have committed to inert the containment during operation to maintain a low level of oxygen. This item is discussed further in Section 22 of the Safety Evaluation Report where we address the rulemaking proceeding for consideration of degraded or melted cores. We conclude that the applicant calculated the hydrogen source in accordance with the provisions of Branch Technical Position CSB 6-2 and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant-Accident."

CONTAINMENT SYSTEMS BRANCH

ISSUE 1

NRC:	No mention in 6.2-32 of debris screens on purge valves.
Supply System:	Commits to provide Seismic Quality Class 1 debris screens on purge valves by fuel load.
	These changes will be made in Amendment 19 to the WNP-2 FSAR
	See revised FSAR page 6.2-65 (attached).

from the control room. These valves provide assurance of isolating these lines in the event of a break and also provide long-term leakage control. In addition, the piping is considered an extension of containment boundary since it must be available for long-term usage following a design basis loss-of-coolant accident, and, as such, is designed to the same quality standards as the primary containment.

6.2.4.3.2.2.3.2 Containment Purge Lines

The drywell and suppression chamber purge lines have isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has two air operated spring closing valves located outside the primary containment. The isolation valves for the purge lines are designed to be locked closed in the main control room. These isolation valves are interlocked to preclude opening of the valves while a containment isolation signal exists as noted in Table 6.2-16.

STAINLESS STEEL GRILLES ARE INSTALLED ACROSS THE SUPPLY AND VENT OPENINGS TO PROHIBIT DEBRIS FROM ENTERING THE PURGE LINES THUS PREVENTING THE ISOLATION VALVES FROM SEATING.

SEISMIC CATEGORY I

6.2.4.3.2.2.3.3 Drywell and Suppression Chamber Air Sampling Lines

The radiation lines penetrate the primary containment and are used for continuously drawing containment air during normal operation as part of the leak detection system. These lines are equipped with two automatic isolation valves located outside and as close as possible to the containment. The hydrogen monitoring lines penetrate the primary containment and are used to continuously monitor the primary containment air during the post-LOCA accident period. These lines are equipped with check or excess flow check valves located outside and as close as possible to the containment. Containment isolation requirements are met on the basis that these lines are low-pressure lines constructed to the same quality standards as the containment and lead to Class 1E essential instruments. Furthermore, the consequences of a break in these lines result in no significant safety consideration.

6.2.4.3.2.2.3.4 Suppression Chamber Spray Lines

The suppression chamber spray lines penetrate the containment to remove energy by condensing steam and cooling noncondensible gases in the suppression chamber. The line is equipped with a normally closed motor-operated valve located outside and as close as possible to the primary containment. This normally closed valve receives an automatic isolation signal. Containment isolation requirements are met on the basis that the spray header injection lines are normally closed, low pressure lines constructed to the same quality standards as the containment.

that the use of the drywell and suppression pool purge system is acceptable provided that the applicants limit purging to control of containment pressure, inerting and deinerting of the containment and qualifies the valves to the requirements of Branch Technical Position CSB 6-4 as discussed further in Section 22 of the Safety Evaluation Report. It should be noted that the current design does not include a debris screen to protect the valves. The applicants have committed to install the debris screens prior to fuel load.

Conclusion

Based on the above evaluation, we conclude that the applicants' proposed design of the containment isolation system satisfies the requirements of Criteria 54, 55, 56 and 57 of the General Design Criteria and is acceptable except for the recombiner scrubber return line to the suppression pool where we will require the applicant to provide redundant isolation valve to meet GDC 56.

6.2.5 Combustible Gas Control

The combustible gas control systems include piping, valves, and components and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentration of these gases.

The scope of our review of the design and functional capability of the combustible gas control system for WPPSS Nuclear Project No. 2 included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentration, and reduce combustible gas concentrations within the containment following the design basis accident. Our review also included the applicants' proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the system provided to support the adequacy of the design bases.

The bases for our acceptance are the conformance of system design and design bases to the Commission's regulations as set forth in the General Design Criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards.

Following a loss-of-coolant accident, hydrogen may accumulate within the containment as a result of corrosion of metals in the containment, metal-water reaction between the fuel cladding, and as a result of radiolytic decomposition of the post-accident emergency cooling water. The applicants analyzed the production and accumulation of hydrogen from the above sources in accordance with the guidelines of Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment." The guideline regarding the metal-water reaction states that hydrogen production is five times the maximum calculated reaction under 10 CFR 50.46, or that amount that would be evolved from a core-wide average depth of reaction into the original cladding of 0.23 mils, whichever is greater, in two minutes. The applicants have committed to inert the containment during operation to maintain a low level of oxygen. This item is discussed further in Section 22 of the Safety Evaluation Report where we address the rulemaking proceeding for consideration of degraded or melted cores. We conclude that the applicant calculated the hydrogen source in accordance with the provisions of Branch Technical Position CSB 6-2 and Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant-Accident."

6.2.4.3 Containment Purge System (CEB-4)

As shown on FSAR Table 6.2-16 and explained in Note 18 (pg. 6.2-139), single containment isolation valve protection is provided for the drain lines from the Containment Atmosphere Control (CAC) System loops A and B (valve no.'s RHR-V-134A and B, respectively). These lines connect to an engineered safety feature system (the CAC System) for which a single isolation valve is acceptable as stated in SRP 6.2.4, Section II, para. 6.e. The acceptability of a single isolation valve is contingent upon all of the following prerequisites:

1) System reliability is improved with only one isolation valve in the line.

2) The system is closed outside containment and a single active failure can be accommodated with only one isolation valve.

3) This closed system is protected from missiles.

4) The closed system is designed to Seismic Category I, Safety Class 2*, and a minimum temperature and pressure rating at least equal to that for the containment.

5) The piping between the isolation valve and containment is enclosed in a leak-tight housing or conservative design of the piping and valve, conforming to SRP 3.6.2, precludes a breach of piping integrity.

The isolation valves for the CAC System loop drain lines meet each of the above criteria. In lieu of a leak-tight housing between the isolation valve and containment, this piping has been designed to comply with SRP 3.6.2, Branch Technical Position MEB-1, Section B.2.b for moderate-energy piping.

* Figure 3.2-17 shows the Standby Service Water (SW) supply to each scrubber in the CAC System as being Safety Class 3 (Code Group C). This is incorrect and Figure 3.2-17 is being modified. All skid-mounted service water piping and valves for the scrubbers are Safety Class 2 (Code Group B). The code group boundary is located on the SW side of valve CAC-V-59A/B. Isolation between the CAC System and the SW System is provided by valve CAC-FCV-5A/B, which closes automatically when its associated hydrogen recombiner is shutdown.

DSER 6.2.6 (CSB-21,22).

The combined leakage from all these valves will satisfy the acceptance criteria of 10 CFR Part 100 regarding the site radiological safety analysis and will be included in the plant technical specifications. This leakage will be excluded when determining the combined leakage rate for all penetrations and valves as specified in Paragraph III.C.3 of Appendix J.

We reviewed the applicants' proposed hydrostatic testing and concluded that such testing is permissible for the lines identified above since the applicants have shown the:

(1) Existence of a water seal,

(2) System boundaries are designed to engineered safety feature criteria, and

(3) Acceptance criteria of 10 CFR Part 100 are satisfied.

Traversing Incore Probe System

The traversing incore probe system is equipped with ball valves that provide the guide tubes with shutoff capability following the cable withdrawal. A shear valve is provided to cut the cable in the event that the drive cable cannot be withdrawn.

The applicants have committed to perform a Type C test on the ball valve. Since the shear valve requires testing to destruction, the applicants are not going to perform periodic Type C tests on these valves. However, statistically chosen samples of the shear valves is tested by the manufacturer. To assure that the shear valve will perform its intended function, we have requested, and the applicants have agreed, to:

(1) Verify the continuity of the explosive charge at least once per 31 days.

(2) Initiate one of the explosive squibs charge at least once per 18 months. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired.

(3) All charges should be replaced according to the manufacturer recommended life time.

Based on the above discussion, we conclude that the leak testing of the traversing incore probe system is acceptable.

Control Rod Drive

Appendix J to 10 CFR Part 50 (see Paragraph III.C.3.(a)) allows exclusion (from combined $0.6L_a$) of leakage from valves that are sealed with fluid from a seal system if the fluid leakage rates do not exceed those specified in the technical specifications.

To assure that system leakage will not exceed 3 gallons per minute (maximum leakage), a piping integrity test is accomplished for leaks of the hydraulic control units (operating pressure 1000 pounds per square inch) during daily

ISSUE 21

NRC: TIP System - Will the explosive valves be bench tested?

Supply System: WNP-2 will review the standard technical specification commitment for bench testing this valve, and inform F. Eltawila of the Supply System position prior to September 25, 1981.

Supplementary Information: The Supply System commits to verify the continuity of the electrical circuitry to the explosive charge in the TIP System, at least once every 31 days, and to initiate one of the explosive squib charges at least once every 18 months. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired, or from another batch which has been certified by having one of that batch successfully fired. All charges will be replaced according to the manufacturer's recommended lifetime.

See revised FSAR pages provided in response to CSB-22.

CONTAINMENT SYSTEMS BRANCH

ISSUE 22

NRC:	Page 6.2-143, TIP system. Response to Question 022.082 shows a Type C test for this valve, yet the test is not indicated in the text of the FSAR.
Supply System:	The Supply System will revise the FSAR text to indicate a Type C test for the TIP ball valve.
	See revised FSAR pages 6.2-141, 6.2-142, and 6.2-143 (attached).

TABLE 6.2-16 (Continued)

a. do not communicate with either the reactor coolant system or the containment atmosphere,

b. are protected against missiles and pipe whip,

c. are designed to withstand temperatures at least equal to the containment design temperature,

d. are designed to withstand the external pressure from the containment structural acceptance test, and

e. are designed to withstand the loss-of-coolant accident transient and environment.

Even if the failed closed valve were to not shut there will be no leakage of containment atmosphere through the hydraulic control lines since the piping inside the primary containment remains intact. There are no active component failures which would compromise the integrity of the closed system inside the primary containment. Integrity of the closed system inside the primary containment is, essentially, constantly monitored since the system is under a constant operating pressure of 1800 psig. Any leakage through this system would be noticed because operation would be erratic and because of indications provided on the hydraulic control unit. In addition, in order to perform type C tests on these lines, the system would have to be disabled and drained of the corrosive hydraulic fluid. This is considered to be detrimental to the proper operation of the system in that possible damage could occur in establishing the test condition or restoring the system to normal.

These lines and associated isolation valves should therefore be considered to be exempt from type C testing.

29. Since the traversing incore probe (TIP) system lines do not communicate freely with the containment atmosphere or the reactor coolant, General Design Criteria 55 and 56 are not directly applicable to this specific class of lines. The basis to which these lines are designed is more closely described by General Design Criterion 54, which states in effect that isolation capability of a system should be commensurate with the safety importance of that isolation. ~~Furthermore, even though the failure of the TIP system lines presents no safety consideration, the TIP system has redundant isolation capabilities.~~

TABLE 6.2-16 (Continued)

The safety features have been reviewed by the NRC for BWR/4 (Duane Arnold), BWR/5 (Nine Mile Point) and BWR/6 (GESSAR), and it was concluded that the design of the containment isolation system meets the objectives and intent of the General Design Criteria.

Isolation is accomplished by a seismically qualified solenoid-operated ball valve, which is normally closed. To ensure isolation capability, an explosive shear valve is installed in each line. Upon receipt of a signal (manually initiated by the operator), this explosive valve will shear the TIP cable and seal the guide tube.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of five valves may be opened at any one time to conduct calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes a cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. If a TIP cable fails to withdraw or a ball valve fails to close, the explosive shear valve is actuated. The ball valve position is indicated in the control room.

INSERT

The ~~WNP-2~~ TIP system design specifications require that the maximum leakage rate of the ball and shear valves shall be in accordance with the Manufacturer's Standardization Society (Hydrostatic Testing of Valves). The ball valves are 100% leak tested to the following criteria by the manufacturer:

Pressure	0 - 62 psig
Temperature	340°F
Leak Rate	10^{-3} cm^3/sec

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

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

ALL THIS INFO IS NOW UNNECESSARY. IT WAS ORIGINAL[LY] MEANT TO SUPPORT THE POSIT[ION] OF NOT TESTING THE TI[P] BALL VALVES

Insert to Page 6.2-142

The ball valves will be type C tested in accordance with Appendix J of 10CFR50. Because the shear valves have explosive squibs and require testing to destruction they will not be type C tested. To assure shear valve operability, however:

1. The continuity of the explosive charge will be verified at least once every 31 days.

2. One of the explosive squib charges will be initiated at least once every 18 months. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one of that batch successfully fired.

3. All shear valve charges will be replaced according to the manufacturer's recommended lifetime.

TABLE 6.2-16 (Continued)

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

As stated above, the penetration is automatically closed following use. During normal operation the penetration will be open appproximately eight hours per month to obtain TIP information. If a failure occurred such as not being able to withdraw the TIP cable, the shear valve could be closed to isolate the penetrations. Installation requirements are that the guide tube/penetration flange/ball and shear valve composite assemble not leak at a rate greater than 10^{-4} std cc/sec at 80 psig. Further leak testing of the shear valves is not recommended since destructive testing would be required.

Leak testing of the ball valves also is not recommended since the guide tube terminates in a sealed indexer housing which is kept under a positive pressure by a nitrogen or air purge. The purge make-up will be indicative of the system leakage. Note that the TIP ball valve is normally closed and thus is a part of the leakage barrier being monitored. Consequently, the personnel exposure required to conduct type C tests from inside the containment is not warranted.

30. System is initiated after a LOCA. Isolation valves will automatically close on the following high leakage conditions:

    a. Five psi between main steam isolation valves, 60 seconds after system initiation

    b. High flow from main steam line to low pressure manifold, 150 seconds after system initiation

    c. Inboard main steam isolation valve opened, after system initiation

31. PCRVIS is not desirable since the feedwater system, although not an ESF system, could be a significant source of make-up after a LOCA which is not concurrent with a seismic event.

    Feedwater check valves on either side of the containment provide immediate leak isolation, if required. The feedwater block valves can, however, be remote-manually closed if there is no indiction of feedwater flow (see 6.2.4.3.2.1.1.1).

DSER 6.3

WNP-2 OPEN ITEMS

1. Overpressurization Protection (5.2.2)- The applicant must submit for our reivew and approval, a plant specific overpressurization analysis using the ODYN code and including the effect of recirculation pump trip.

2. Safety/Relief Valve Surveillance (5.2.2)- The applicant must commit to participate in a surveillance program to monitor the performance of safety/relief valves.

3. Pressure Interlocks on ECC Injection Valves (6.3)- The applicant must verify that interlocks are present at all times for both manual and automatic valve operation and that the interlocks do not allow valve opening until the reactor coolant pressure is below the design pressure of the ECC system involved, or provide an alternative configuration which satifies the requirements of SRP Section 6.3.

4. Premature LPCI Diversion (6.3)- The applicant must provide assurance that LPCI flow will not be diverted to containment cooling before adequate core cooling is provided. (We have accepted a discussion of emergency procedures and operator training for this item on other applications.)

5. Long Term Air Supply to ADS Valves (6.3)- The applicant must verify that the bottled air supply serving as a backup to the normal air supply for the ADS valves is valved in during normal operation, or provide justification as to why credit should be given to this air supply.

6. Thermal Power Monitor in Transient Analyses (15)- We require that the thermal power monitor time constant be included in the plant technical specifications or that no credit be taken for the thermal power monitor in transient analyses.

6.3 Premature LPCI Diversion (RSB4)

Question:

Provide assurance that LPCI flow will not be diverted to containment cooling before adequate core cooling is provided.

Response:

The LPCI (low pressure coolant injection) mode is the design basis mode for the RHR (residual heat removal) system in ICC (emergency core cooling) operation. The LPCI mode is initiated automatically on low coolant level or high containment atmosphere pressure. Any other mode of RHR operation which might be in progress at the time is automatically valved off. For the postulated worst case, which is the instantaneous double-ended gillotine pipe rupture. All three RHR pumps are started automatically in the LPCI mode to flood the reactor core. After level has been restored and the RPV has depressurized, one RHR pump operating in the LPCI mode is sufficient for core cooling. Loop A or Loop B (which include a heat exchanger) or both may then be used for containment spray or suppression pool cooling. The operator can not open containment spray or suppression pool cooling valves until the injection valve in the same loop is fully closed; since RPV level change is the expected response to changing injection valve position, the operator's attention will be directed to the operating parameter ensuring adequate core cooling.

Additionally, emergency procedures now being developed using the BWR Owner's Group guidelines will caution the operator to assure adequate core cooling prior to diverting injection flow away from the core.

6.3 Long Term Air Supply to ADS Valves

Question:

Verify that the bottled air supply for the ADS valves is valved in during normal operation, or provide justification as to why credit should be given to this air supply.

Answer:

During normal operation, all eighteen relief valves are supplied with pressurized air from a common supply header. In addition, each valve has an individual air accumulator with an inlet check valve. Each valve can continue to operate for a number of strokes from the accumulator, if air pressure is lost in the header. Seven of these 18 valves are used for the ADS (automatic depressurization system) function. In addition to the regular air accumulator for these seven valves, each valve has a redundant pilot valve with a corresponding redundant accumulator and check valve. These redundant accumulators are supplied from two nitrogen supply headers which during normal operation are pressurized with air. If air pressure is lost, pressurized nitrogen is supplied to the headers.

The nitrogen supply header from bottle bank A supplies three ADS valves. Bottle bank B supplies four ADS valves. If pressurized air is lost, each nitrogen supply header is individually isolated from the common air supply line once the nitrogen pressure closes the corresponding check valve.

Nitrogen supply is initiated when pressure sensor CIA-PS-21A (for bottle bank A) or CIA-PS-21B (for bottle bank B) senses a line pressure below 140 psig. All nitrogen bottles are valved into the system during normal operation, except that a solenoid valve on each individual bottle is normally closed. The low pressure signal from the air pressure sensor activates the stepping programmer (CIA-PROG-1A for bank A, CIA-PROG-1B for bank B) which open the solenoid valve for the first nitrogen bottle. This bottle then supplies nitrogen through a pressure reducing valve to the nitrogen supply header and redundant nitrogen accumulator. After the first gas bottle is depleted, the stepping programmer is again activated by a low pressure signal. The programmer advances by one step, opens (de-energizes) the valve for the depleted bottle. In this way, the two banks of bottles are controlled independently and each bank uses only one bottle at a time. Plant startup check lists will include a step to assure the CIA system is properly aligned to support plant operation.

References

Burns & Roe P&ID M-556

Burns & Roe System Description 24

Burns & Roe Logic Diagram on drawing M-620, sheets 556-5 and 556-6

Dser 6.3.2.3

One of the design requirements of the emergency core cooling system is that cooling water flow be provided rapidly fowlloing the initiation signal. By always keeping the emergency core cooling system pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the reactor pressure vessel can be minimized. In addition, full discharge lines will prevent potentially damaging water-hammer occurrences on system startup. At WNP-2 a fill system consisting of a jockey pump in the RCIC system and in each of the ECCS subsystems (except ADS) is provided. Maintenance of the filled status of the system is ensured by continuous indication of pump operation and pump discharge pressure. In accordance with monthly surveillance procedures the vent lines in the filled systems are opened and checked for flow to eliminate the possibility of the formation of air pockets. Pressure instrumentation provided on the jockey pump discharge line initiates an alarm in the main control room when pressure in the discharge line is less than the hydrostatic head required to maintain the line full of water up to the injection valves.

1 VERIFY

2 VERIFY

The emergency core cooling system pumps must have the capability to operate for an extended period of time during the long-term recirculating cooling phase following a loss-of-coolant accident. The applicant has provided pump reliability information based upon actual operation experience on similar pumps manufactured by Ingersall-Rand, the WNP-2 supplier. As discussed in Section 5.4.7 we are awaiting additional information from the applicant on deep draft pump reliability.

Safety/relief valve operability will be demonstrated during the power ascension phase of the plant startup test program by manually actuating each safety/relief valve (including the ADS valves) one at a time to measure discharge capacity and to demonstrate that no blockage exists in the valve discharge line. After commercial turnover all of the safety relief valves will be tested in accordance with Section XI, Article IWV of the ASME Boiler and Pressure Vessel Code. The applicant has also stated that direct valve position indication, via acoustic

SEC 6.3.1

The applicant also discussed the simultaneous closure of a recirculation flow control valve during a loss-of-coolant accident. The applicant's basis for the closure time was an electronic velocity limiter designed to limit the opening and closing rate to 11 percent per second. The valve controller is not classified as equipment which is essential to safety. Therefore, the controller is not scrutinized by us to the same extent that a component required for safety would be. However, for the loss-of-coolant accident, the valve is not called upon to actively mitigate the consequences of the accident, but is only needed to passively remain in its current position. If the control valve were in the automatic mode at the time of the loss-of-coolant accident, the control system would normally call for the valve to open. If the control valve were in the manual mode, operator action would normally be required to close the valve. Two failures in the control logic would be required for the valve to close even at the 11 percent per second rate. Further, the control logic is outside the drywell and is not subject to the loss-of-coolant accident environment. Therefore, the 11 percent per second closure rate, which will be verified by periodic tests required by plant technical specifications, is a reasonable limit on valve closure rate for the loss-of-coolant accident analysis. The applicant indicated that if this additional failure is included in the loss-of-coolant accident, the peak temperature for the worst break would be increased approximately 50 degrees Fahrenheit which still satisfies the criterion of 10 CFR Section 50.46. Therefore, we find the loss-of-coolant accident analyses to be acceptable.

3

VERIFY

Insert B2+ [illegible]

The design of the SGTS was compared to the criteria of Regulatory Guide 1.52 (1) and the following deviations were noted:

Section 2g: The SGTS system is not instrumented to record pertinent pressure drops and flow rates at the control room. Instrumentation is provided only to alarm abnormal pressure drops, to record alarms (abnormal pressure drop), and to indicate (but not record) system flow rate. The applicant's FSAR notes these deviations but gives no bases or justifications for the deviations. ~~Licensee~~ Applicant should provide his rationale for not meeting Section 2g.

Section 2i: The FSAR states that SGTS filters can be bypassed for testing but that there is no provision for indication of bypass status in control room (Note 4, page 6.5-20); in effect, this negates the automatic activation of these systems in the event of an accident if the bypass is in use and does not keep the operator informed of system status.

Section 3a: Note 6, page 6.5-20 of the FSAR, indicates that the SGTS demisters are not qualified in accordance with the requirements of MSAR 71-45, but that the manufacturer contends that the addition

SER Items Requiring Verification

1 & 2* Verification that the ECCS discharge line fill systems are provided with continuous indication of pump operation, pump discharge pressure and low pressure alarms in the control room is provided in the response to question 211.079 submitted in Amendment 11, September 1980.

3.** Verification that the RRC flow control valve closure time is limited to an opening and closing rate of 11% per second by a flow limiter is provided in response to question 211.188 submitted in Amendment 20, November, 1981.

Verification that the RRC control valve would tend to open post-LOCA when in the automatic mode is provided in response to question 31.001(3) submitted in Amendment 14, April 1981, and question 31.058, submitted in Amendment 3, March 1979.

Verification that the maximum peak cladding temperature for the worst break would be increased by approximately 50°F (≐45°F) is provided in the response to question 211.083, submitted in Amendment 11, September 1980.

4.*** See response to SER Open Item 6.3 (RSB-4).

* - Draft SER Section 6.3.2.3, 5th paragraph.
** - Draft SER Section 6.3.4, 7th paragraph.
*** - Draft SER Section 6.3.4, 8th paragraph.

The low pressure coolant injection flow may be diverted manually to drywell spray cooling or to suppression pool cooling. The applicants

4

have stated that the WNP-2 emergency procedures contain adequate cautions to deter the operator from premature flow diversion. These procedures, which are based on guidelines accepted by us (see Chapter 22 Item I.C.1), caution the operator against diversion unless adequate core cooling is assured. LPCI diversion is identified in the procedure as secondary to core cooling requirements except in those instances outside the design envelope involving multiple failures for which maintenance of containment integrity is required to minimize risk to the environment. We have reviewed the containment response analyses for the design basis event to determine the need for low pressure coolant injection diversion. These analyses indicate that there should be no need for wetwell spray actuation in the time frame during which the peak cladding temperature is reached. The operator's focus would, therefore, be on maintaining core cooling. Based on these analyses and the emergency procedures discussed above, we find the applicants' position on low pressure coolant injection diversion to be acceptable. Review of all emergency procedures is being addressed in Items I.C.1 and I.C.8 of Chapter 22 of this report.

VERIFY

The core spray sparger for both the high and low pressure core spray systems each consists of two semicircular segments which form an essentially complete circular sparger. Water is sprayed radially onto the tops of the fuel assemblies by short elbow nozzles spaced around the sparger. Tests of this type of spray system were performed in a full-scale test in which air at atmospheric pressure simulated the post loss-of-coolant accident steam environment and indicated adequate cooling was delivered to each fuel assembly. However, recent tests conducted on a single nozzle indicate that the actual steam environment may adversely affect the distribution of flow from certain types of core spray nozzles. As discussed in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," this problem is being studied by us under Task Action A-16 entitled, "Steam Effects on BWR Core Spray Distribution." Preliminary analyses and measurements have been made which support the existence of a significant safety margin between that

Insert B2 + [illegible]

The design of the SGTS was compared to the criteria of Regulatory Guide 1.52 (1) and the following deviations were noted:

Section 2g: The SGTS system is not instrumented to record pertinent pressure drops and flow rates at the control room. Instrumentation is provided only to alarm abnormal pressure drops, to record alarms (abnormal pressure drop), and to indicate (but not record) system flow rate. The applicant's FSAR notes these deviations but gives no bases or justifications for the deviations. ~~Licensee~~ Applicant should provide his rationale for not meeting Section 2g.

Section 2i: The FSAR states that SGTS filters can be bypassed for testing but that there is no provision for indication of bypass status in control room (Note 4, page 6.5-20); in effect, this negates the automatic activation of these systems in the event of an accident if the bypass is in use and does not keep the operator informed of system status.

Section 3a: Note 6, page 6.5-20 of the FSAR, indicates that the SGTS demisters are not qualified in accordance with the requirements of MSAR 71-45, but that the manufacturer contends that the addition

of 3 two-inch fiberglass ~~pads~~ pads "should be approved". ~~Licensee~~ Applicant should verify the current status of approval and state any pertinent reasons as to why approval has not been obtained.

Section 4d: Section 4d of Regulatory Guide 1.52 recommends that each ESF atmosphere cleanup train be operated at least 10 hours per, month, with heaters on, in order to reduce the buildup of moisture on the adsorber and HEPA filters. The purpose of this requirement is the removal of accumulated moisture from the HEPA filter and carbon adsorber components. ~~Licensee~~ Applicant, in Note 13, page 6.5-22 of the FSAR, maintains that periodic activation of strip heaters is adequate to maintain the charcoal beds moisture-free and that simultaneous operation of the fans is not required. ~~Licensee~~ Applicant should provide details of any test data which substantiates his position; otherwise, the Technical Specifications will be conditioned to require operation of each ESF train for at least 10 hours per month, with heaters on, to reduce moisture buildup.

Testing of ESF atmosphere cleanup system components was reviwed with respect to the criteria of Regulatory Guide 1.52. Test provisions were in accordance with the criteria with the following exceptions:

- Inplace testing of HEPA filter section is in accordance with Military Standard MIL-STD-282, which is incorporated by reference in ANSI N510; however, the listed testing criteria do not make mention of those portions of Position C.5.b of

Regulatory Guide 1.52, which provide for testing at specific intervals or following painting, fire, or chemical release (this may be an open item or, alternatively, may be in the Tech Specs). To be in conformance with Regulatory Guide 1.52, applicant should also reference MIL-F-51068 for testing of HEPA filters.

OPEN ITEM: Does not mention testing requirements for specific intervals between tests and does not specify testing following painting, fire, or chemical release. TEsting of HEPA filters should reference MIL-F-50168; alternatively, ANSI-N510 could be referenced, which would then incorporate references to both MIL-STD-282 and MIL-F-51068.

Instrumentation requirements for the SGTS were reviewed with respect to the criteria of Regulatory Guide 1.52. Instrumentation provisions were in accordance with the criteria with the exceptions noted below.

Instrumentation provided for the STGS includes:

- Flow rate, unit outlet. Provides flow rate indication and high/low Δp alarms in the main control room. No provisions for recording of flow as recommended in ANSI 509 and Regulatory Guide 1.52.
- ~~Temperature, charcoal bed. Provides high alarms in the main control room but no direct indication of temperature. (No requirements specified in Regulatory Guide 1.52)~~

~~6.5.2.1~~ - Pressure Drop (Δp). For each element in the SGTS trains, system provides indication, status of operation and high Δp alarms in the main control room. Components covered include roughing filter, upstream HEPA filter, charcoal beds, and downstream HEPA filter. No provision noted for recording of any system pressure drops. Alarms of high Δp are recorded in the plant computer. No provisions for measurement of total pressure drop across complete system. Regulatory Guide 1.52, Section C.2.g, recommends recording of "pertinent" pressure drops at the control room.

OPEN ITEMS: No provision for control room recording of system air flow rates or pressure drops. No provision for measurement, indication, or recording of total system pressure drop (ΣΔp). No provision for status indication in control room of deluge valve positions, valve/damper operator position, or fan status.

6.5.1.2.2
~~6.5.2.2~~ Control Room Emergency Filter System (CREFS)

The function of the control room emergency filter system (CREFS) is to supply non-radioactive air to the control room after a DBA and to pressurize the control room. This system will permit operating personnel to remain in the control room following a DBA. The CREFS is a redundant system, with each system having an intake design capacity of up to 1,000 cfm of air and recirculating design capacity of 1,000 cfm of air. Each system contains the following components:

Open Issue: Chapter 6.5.1.2.1
Section 2g of Regulatory Guide 1.52

The SGTS system is not instrumented to record pertinent pressure drops and flow rates at the control room. Instrumentation is provided only to alarm abnormal pressure drops, to record alarms (abnormal pressure drop), and to indicate (but not record) system flow rate. The applicant's FSAR notes these deviations but gives no bases or justification for the deviations. Applicant should provide his rationale for not meeting Section 2g.

Response:

See revised Table 6.5-2, Note 3.

Open SER issue chapter- 6.5.1.2.1
Section 2i of Regulatory Guide 1.52

The FSAR states that SGTS filters can be bypassed for testing but that there is no provision for indication of bypass status in control room (Note 4, page 6.5-20); in effect, this negates the automatic activation of these systems in the event of an accident if the bypass is in use and does not keep the operator informed of system status.

Response:

Note 4 on page 6.5-20 was meant to address item C-2-j of RG1.52, not C-2i. The FSAR has been revised to correct this error. (draft FSAR page change attached)

However, in direct response to the comment in the draft SER, the SGTS is in full compliance with RG1-47. This is discussed in FSAR Section 7.1.2.4. The logic diagrams (Fig. 7.3-14) are in the process of being updated to show the system bypass indication logic.

If the system is bypassed mechanically for filter replacement, procedure would call for locking the isolation dampers to the appropriate filter bank closed. This would automatically annunciate a system inoperable status in the control room. In addition, procedure would call for manual initiation of system bypass status indication for any maintenance operation which might render a filter train inoperative.

Open SER Issue 6.5.1.2.1
Section 3a of Regulatory Guide 1.52

Note 5, page 6.5-20 of the FSAR, indicates that the SGTS demisters are not qualified in accordance with the requirements of MSAR 71-45, but that the manufacturer contends that the addition of 3 two-inch fiberglass pads "should be approved". Applicant should verify the current status of approval and state any pertinent reasons as to why approval has not been obtained.

Response :

See revised Table 6.5-2, Note 5.

Open SER Issue 6.5.1.2.1
Section 3a of Regulatory Guide 1.52

Note 5, page 6.5-20 of the FSAR, indicates that the SGTS demisters are not qualified in accordance with the requirements of MSAR 71-45, but that the manufacturer contends that the addition of 3 two-inch fiberglass pads "should be approved". Applicant should verify the current status of approval and state any pertinent reasons as to why approval has not been obtained.

Response :

See revised Table 6.5-2, Note 5.

TABLE 6.5-2 (Continued) Page 3 of 5

Note 1
C-2.a

Demisters are not provided in the control room filter units due to the absence of entrained moisture during normal and abnormal conditions. HEPA filters are not provided after the charcoal filter because filter unit discharges into control room air conditioning unit on intake side of medium efficiency filters.

Note 2
C-2.d

Both units of the standby gas treatment system are located in secondary containment and are not subject to containment pressure surges during accidents. Redundant Seismic Category I valves in series isolate and protect these units from containment DBA pressures. Both units of the control room emergency filter system are not subject to containment pressure surges during accidents.

Note 3
C-2.g

INSERT(1)

~~Abnormal pressure drops across all critical components of the SGTS and control room filter units cause an alarm in the main control room and flow through the SGTS is indicated in control room; however, no facilities to record these readings are provided.~~

~~Computer input is provided to record high pressure alarms across critical components and low flow at discharge from SGTS fans.~~

Note 4
C-2.~~i~~j

SGTS filter units are not designed to be removable from the building as an intact unit. The size of the units precludes removal in one section. In the event the units become radioactively contaminated they will be permitted to decay in place until radiation levels are sufficiently low to permit the removal of all internals for disposal.

Note 5
C-3.a

INSERT(2)

~~Farr demisters without fiberglass pads do not meet qualification requirements of MSAR 71-45 due to~~

TABLE 6.5-2 (Continued) Page 4 of 5 INSERT(2)

~~"visible" carry-over. Farr contends that by adding 3 two inch fiberglass pads their demister (which was acceptable in the Fort St. Vrain clean up filter housing) should also be approved. There are no demisters on the control room filter units See Note 1 (C-2.a) above.~~

Note 6
C-3.d HEPA filters are not subjected to iodine removal sprays; therefore, aluminum separators are used.

Note 7
C-3.g Access doors into SGTS units are 50 x 20 inches. Vacuum breakers are not provided on doors of SGTS and control room units. Unit fans are normally off. During tests, bypass is via temporary blanking off of doors.

Note 8
C-3.i <u>Test 4, Activity</u>

Base carbon (unimpregnated) activity test was not previously required and because all available carbon was of the impregnated type this test was not run.

<u>Test 5a, Radioiodine Removal Efficiency</u>

The activated carbon (Barnebey Cheney 727) radioiodine removal efficiency for methyl iodide at 25°C and 95% relative humidity is 98% instead of 99%.

The methyl iodide test at 25°C and 95% R.H. was not required by Regulatory Guide 1.52, Rev. 0, or contract specification. Considering also that the adsorber bed will not see any relative humidity above 70% the test results of 98% removal efficiency at 95% R.H. should be satisfactory.

Note 9
C-4.a Doors provided on SGTS units are 50 x 20 inches. Access panels are provided on control room units. Vacuum breakers are not provided on any of the units since they are normally not operational.

Note 10
C-4.b Control room filter units have approximately 18 inches between prefilter and HEPA filter frames, and approximately four feet are provided between HEPA and charcoal filter frames. SGTS filter units have a minimum of three feet provided between demister, heater, prefilter, HEPA, and charcoal filter frames.

Insert (1) to Page 6.5-20

Note~~s~~ 3

C-2.g

Abnormal pressure drops across critical components of the SGTS and control room filter units cause an alarm in the main control room, however no facilities to record the pressure drops are provided. A record of pressure drop across individual components and the total SGTS system would be of no value because the SGTS is a variable flow system, with flow modulated to maintain the reactor building at a fixed negative pressure. Flow through the system, which is the pertinent parameter, is recorded in the main control room, and computer input is provided to record high pressure alarms across critical components.

Insert (2) to Page 6.5-20:

Note 5, C-3.a

SGTS system demisters furnished by FARR company, are not in complete conformance with ANSI 509-1976 because they were not qualified by testing in accordance with USAEC report MSAR-71-45. A Moisture Eliminator Study performed by FARR company in 1970, which did not conform to the MSAR-71-45 test set-up, indicated that the installed demisters will protect the HEPA filters in the system from blinding under conditions far more severe than those hypothesized for the SGTS system. Since, under the accident mode, entrained water droplets will not be in the inlet air stream, the FARR tests and qualification are considered adequate.

Open SER Issue

6.5.1.2 Section 4d of Regulatory Guide 1.52

Section 4d of Regulatory Guide 1.52 recommends that each ESF atmosphere cleanup train be operated at least 10 hours per month with heaters on. Applicant, in Note 13, p. 6.5-22 of the FSAR, maintains that periodic activation of strip heaters is adequate to maintain charcoal beds moisture-free and that simultaneous operation of the fans is not required. Applicant should provide details of any test data which substantiates this position; otherwise, the Technical Specifications will be conditioned to require the operation of each ESF train for at least 10 hr per month, with heaters on, to reduce moisture buildup.

Response

Technical Specification 4.6.5.3.a states that each SGTS subsystem shall be operated for at least 10 hr/month with the heaters in a moisture reducing mode.

Open SER Issue 6.5.1.2.1

Review Position With Regard To Reg. Guide 1.52, C.5.b

Listed testing criteria do not make mention of Position c.5.b of Reg. Guide 1.52.

Response

Initial testing of airflow distribution to HEPA filters and iodine adsorbers will be included with preoperational testing of the SGTS. Requirements for testing after maintenance affecting the flow distribution are included in the Technical Specifications, Section 4.6.5.3.

Open SER Issue 6.5.1.2.1

Testing Following Painting, Fire, or Chemical Release

Does not mention testing requirements for specific intervals between tests and does not specify testing following painting, fire, or chemical release. Testing of HEPA filters should reference MIL-F-50168; alternatively, ANSI N510 could be referenced, which would then incorporate references to both MIL-STD-282 and MIL-F-51068.

Response

Technical Specification 4.6.5.3.b specifies testing of the SGTS at least once per 18 months or following painting, fire, or chemical release, in accordance with Reg. Guide 1.52, Section C.5.2. Reference to ANSI N510 is also included in Technical Specification 4.6.5.3.b.

SER Open Issue 6.5.1.2.1
Instrumentation for SGTS

Flow rate, unit outlet. Provides flow rate indication and high/low Δp alarms in the main control room. No provisions for recording of flow as recommended in ANSI 509 and Regulatory Guide 1.52.

Response:

The flow rate recording for the SGTS has been added to the WNP-2 design. Please see revised 6.5.1.2. (attached) *

*Note: See response to SER issue 6.5.1.2.1 on Section C.2.g of RG 1.52 for rationale on Δp.

SER Open Issue 6.5.1.2.1
Section C.2.g of Regulatory Guide 1.52

Pressure Drop (Δp). For each element in the SGTS trains, system provides indication, status of operation and high Δp alarms in the main control room. Components covered include roughing filter, upstream HEPA filter, charcoal beds, and downstream HEPA filter. No provision noted for recording of any system pressure drops. Alarms of high Δp are recorded in the plant computer, No provisions for measurement of total pressure drop across complete system. Regulatory Guide 1.52, Section C.2.g, recommends recording of "pertinent" pressure drops at the control room.

Response:

The SGTS system is a variable flow system with flow modulated to maintain the reactor building at a fixed pressure with respect to outdoors (-.25 inch W.G.). Since system flow is a variable, pressure drop across the system filters, which is proportional to the square of the flow, is a meaningless variable except during tests when the unit is operated at the design flow rate of 4,457 cfm. Therefore, the pressure drop (Δp) recorders and total system pressure drop recorders are not being added because this data would be of no value for this system.

With regard to indication of system status in main control room, please note the following:

a. The fire protection system deluge valve assembly is presently being revised to a supervised system. Status and fault annunciators will be added to the main control room fire display panel.

b. Status of the SGTS system (system unoperable) is annunciated on status annunciator board BD-S in the main control room. Misalignment of any valve or fan in the system automatically causes inoperable status display in accordance with Regulatory Guide 1.47. FSAR Figures 7-3-19 are being revised to indicate the present system. During system testing, the misalignment of fan discharge valves will result in system inoperable status display annunciations.

NOTE: Response to similar SER item on page 20 of SER 6.5.1.2.1 on Item C.2.j revises note 3a addressing Item C.2.g to provide justification.

6.5.1.2

two units. The partition wall, which is of Seismic Category I design, serves as both a missile barrier and fire barrier between the two units.

During normal plant operation, both SGTS units are on standby. In this mode the only portions of the system which is operational are the strip heaters in the filter units, which are cycled on and off by thermostats set to maintain the filter plenum at 90°F to ensure that the relative humidity within the plenums never exceed 70%, thus protecting the charcoal adsorber from condensed moisture.

The maximum dew point temperature in the reactor building during normal plant operation is 75°F. When in standby, all isolation valves downstream of the unit fans are closed.

In the event that a purge of the primary containment is required, but radiation monitors within the containment indicate that radiation levels are too high for direct purge through the reactor building exhaust system, the purge exhaust can be performed through the SGTS. All controls for the SGTS are located in the main control room from where a control room operator starts the SGTS fans and opens all isolation valves between the containment and SGTS and atmosphere. Purge flow rate is RECORDED AND adjusted by means of electronic flow RECORDING controllers mounted in the control room which transmit control signals to the inlet vanes of the SGTS fans. The sensor for each flow RECORDING controller is in the discharge duct of the fan controlled. Purge supply air to the primary containment is supplied from the reactor building supply air system (See 9.4.2). During primary containment purge both SGTS units can be operated if a purge rate greater than 4457 cfm is desired.

Upon completion of containment purge all primary containment isolation valves are closed and the SGTS inlet valve from the secondary containment is opened. Air from the secondary containment is then drawn through the SGTS unit and discharged to atmosphere. This operation is performed at a reduced flow rate (approximately 500 cfm) through a recycle timer to dissipate the decay heat from radioactive contaminants collected in the filters. The recycle timer is capable of energizing the SGTS unit fan for 1 to 15 minutes every 30 minutes to 3 hours. The cooling operation is terminated and the SGTS unit returned to the standby mode when the radiation level at the filters decays to background levels as determined by the use of portable radiation monitors.

Since the system would be called on to function only in the event of a Loss-of-Coolant Accident (LOCA), we have reviewed the system to assure that it is capable of performing its safety function under the expected LOCA environmental conditions appropriate to the system equipment location. Further, the components of such subsystem are protected by separation and barriers against internally generated missiles, externally generated missiles, and dynamic effects associated with pipe breaks such that their function will not be impaired under postulated LOCA conditions, thus the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," and Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," are satisifed.

[The applicant must verify that the leakage rates used are as specified in the Standard Technical Specifications are used before we can confirm compliance with the guidelines of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants."]

Based on our review, we conclude that MSIVLCS is in conformance with the requirements of General Design Criteria 2 and 4 with respect to its protection against natural phenomena, missiles and environmental effects, and the guidelines of Regulatory Guides 1.29, 1.102 and 1.117 and Branch Technical Position ASB 3-1 with respect to its protection against flooding, tornado missiles and pipe break effects, and is, therefore, acceptable. [We cannot conclude that the guidelines of Regulatory Guide 1.96 relating to the MSIVLCS functional

Open SER Issue

6.7 Main Steam Isolation Valve Leakage Control System

The MSIV-LCS, as designed, imposes a maximum process load of 80 lbs. of saturated steam at 35 psia, vented to the reactor building volume served by the Standby Gas Treatment System (SGTS), followed by the continuous MSIV leakage flow rate of 46 scfh (11.5 scfh at 25 psig per valve).

The initial discharge will have no significant effect on building pressure buildup and the continuous flow is considered negligible compared to the SGTS design flow rate of 4000 scfm. The MSIV-LPS conditions the exhaust temperature and humidity to within the design requirements of the SGTS prior to delivery to the SGTS by diluting the 46 scfh MSIV Leakage with 50 scfm of air from the reactor building.

WNP-2 Technical Specifications, Section 3.6.1.2, limits the MSIV leakage to 11.5 scfh, per valve, at 25 psig and requires testing each valve to verify the leak rate at least once every 18 months. The MSIV leak rates are, therefore, verified to be within the design capability on a routine basis.

Failure of the unit to successfully complete this series of tests will require a review of the system design adequacy and the cause of the failure, and the tests will continue until 128 valid tests are achieved with no more than one failure. This qualifciation program is in conformance with the staff position and is acceptable. The applicant has not yet provided the results of the reliability testing programs for our review. The successful culmination of such a test program provides sufficient bases to conclude that the HPCS diesel generator has the reliability required by General Design Criterion 17. These documentation is expected soon and we will address our review of these results in a supplement to this report.

IEEE 387-1977 Section 5.6.2.2(1) and Regulatory Guide 1.108 position C.1.b.3 recommend that the periodic testing of diesel generator units should not impair the capability of the unit to supply emergency power within the required time. The diesel generator unit design should include an emergency override of the test mode to permit response to bona fide emergency signals and return control of the diesel generator unit to the automatic control system. The override feature has not been provided in the WNP-2 diesel generator unit design, and therefore the system as designed is not acceptable. We will pursue this item with the applicant and provide our results in a supplement to this report.

Branch Technical Position ICSB 17 (PSB) (in Appendix 8A of the Standard Review Plan) requires that diesel generator protective trips be bypassed when the diesel generator is required for a design-basis event. All protective trips are allowed during periodic testing. The allowed exceptions to the above

ITEM Agenda Item #1
(from 9/25/81 Mtg.)

REMARKS: The draft SER for WNP-2 on Chapter 8 (p 13 and 14) states that new and previously untried diesel generator designs are required to undergo a prototype reliability verification testing program. This reliability test is made up of 300 valid start and loading tests with no more than 3 failures. All diesel generators for WNP-2 have successfully passed this test. The acceptability of the HPCS diesel degenerator, however, is conditioned by an additional prototype testing program which consists of 69 successful start and loading tests using the actual generator loads. This program is detailed in NEDO 10905 Rev. 2. As stated in the SER, the 69 start testing program is approved.

The program was applied successfully on a diesel generator at LaSalle County Station. The WNP-2 HPCS diesel generator is considered qualified to the 69 start requirement based on similarity of design and configuration with that of the LaSalle unit. The results of the LaSalle test are documented in NEDO 10905 Rev.3. A comparison of the WNP-2 and LaSalle units is provided in the attached sheet.

RESOLUTION: The NRC staff agrees that the preoperational 69 start test per Reg. Guide 1.108 is an acceptable basis for qualification of the HPCS diesel generator provided the generator is loaded with the HPCS pump at least five times. The Supply System agrees to modify the FSAR to describe the test procedure above.

See revised FSAR pages 8.3-48 and 8.3-48a (attached).

COMPARIZON OF HPCS DIESEL GENERATORS USED IN LA SALLE & WNP-2

COMPONENT	DATA: LA SALLE	DATA: WNP-2
ENGINE:		
MODEL	EMD-20-645E4	EMD-20-645E4
HP	3600	3600
SPEED (RPM)	900	900
GENERATOR:		
MODEL	IDEAL ELEC.-CO-SAB	GE-264X730
KVA	3560	3560
VOLTS	2400/4160Y	2400/4160Y
HERTZ	60	60
INSULATION	CLASS B	CLASS B
MOMENT OF INERTIA (#-FT²)	36,000	32,450
REACTANCE (%)		
SUBTRANSIENT (X''_d)	7.1	10.9
TRANSIENT (X'_d)	14.3	16.5
SYNCHRONOUS (X_d)	112.	82.
TIME CONST (SEC)		
T''_d, (S-C)	.021	.01
T'_d (O-C)	3.5	2.87
REGULATOR TYPE	SOLID STATE	SOLID STATE
EXCITER TYPE	BRUSHLESS ROTARY	STATIC
GOVERNOR:		
MODEL	WOODWARD UG-8	WOODWARD EGB-10
SPEED SENSOR	MECHANICAL	ELECTRONIC WITH MECHANICAL BACKUP
LOADS:		
HPCS PUMP	3000 HP 1800 RPM 373 FLA	3000 HP 1800 RPM 373 FLA
AUX LOADS	220 KW	220 KW

NOTES: (1) WNP-2 OVERALL DIESEL GEN MOMENT OF INERTIA IS LOWER THAN LA SALLE'S. HENCE D-G STARTING TIME MAYBE LESS FOR WNP-2 THAN FOR LA SALLE.

(2) THE HIGHER TRANSIENT & SUBTRANSIENT REACTANCE FOR WNP[illegible] IS COMPENSATED BY THE USE OF A STATIC [illegible] PROVIDES FASTER VOLTAGE RECOVERY DURING VOLTAGE DIPS.

(3) EL[illegible]

3. on a restart with an initial engine temperature equal to the continuous rating, full load engine temperature;

c. Carry the design load for 2000 hours;

d. Maintain voltage and frequency within limits that will not degrade the performance of any of the loads composing the design load below their minimum requirements, including the duration of transients caused by load application or load removal;

e. Withstand any anticipated vibration and overspeed conditions. There is no flywheel coupled with the HPCS diesel generator. The generator and exciter are designed to withstand 25% overspeed without damage.

The HPCS diesel generator has continuous and short-term ratings consistent with the requirements of Section 5.1 of the standard.

Mechanical and electrical system interactions between the HPCS diesel generator unit and other units of the standby power supply, the nuclear plant, the conventional plant, and the Class 1E electrical systems are coordinated so that the HPCS diesel generator units' design function and capability are realized for any design basis event except failure of the HPCS diesel generator unit.

The qualification requirements of IEEE Standard 323-1971 are met by test and on operating experience on similar equipment in similar environment in other plants.

8.3.1.2.2 Tests and Inspection

The auxiliary AC power system is designed to permit periodic testing and inspection of the system as a whole and of the operability and functional performance of the components in accordance with General Design Criterion 18. Preoperational testing, as described in Chapter 14, will be performed to verify that all components, automatic and manual controls, and sequences of operation of the standby power system function as required. Preoperational testing of redundant portions of the onsite electrical power system to verify proper load group assignments is performed in accordance with NRC Regulatory Guide 1.41, Revision 0.

Insert attached

FSAR Addition: (INSERT after page 8.3-48)

Preoperational testing will include demonstrating the required reliability of the WNP-2 standby diesels by means of a start/load test described in Regulatory Guide 1.108. Since Division 1 and 2 diesel-generators are similiar, each will undergo a 35 start/load test. Division 3 (HPCS) will undergo a 69 start/load test.

Description of Division 3 test;

To accomplish this test, supply of 4.16 KV class 1E bus SM-4 will be transferred to the startup source. Under this condition, the diesel-generator will be synchronized to the 230 KV startup source and loaded via manual adjustment of the unit speed controls to at least 50% of continuous rating and operated at this level for at least one hour. 64 such tests will be accomplished. In addition, 5 tests will involve loading the diesel with the existing bus loads including the HPCS pump motor.

Significant parameters such as voltage, frequency, operating temperature, acceleration times and other pertinent functions will be monitored throughout the deration of the test and recorded. Valid tests and failures will be based on the criteria of Regulatory Guide 1.108 section 2.e. Failures considered are limited to those caused by malfunction of the diesel-generator set only. Failures caused by malfunctions in the test equipment, external circuitry, or loads are not considered attributable to the reliability of the diesel-generator set. Provisions are made to determine the cause of any malfunction or excess wear and to classify it as a valid failure of the equipment being tested or an external nonvalid failure. Such determination of cause and classification of failure will be fully suported by documentation.

Failure of the unit to successfully complete this series of tests will require a review of the system design adequacy and the cause of the failure, and the tests will continue until 128 valid tests are achieved with no more than one failure. This qualifciation program is in conformance with the staff position and is acceptable. The applicant has not yet provided the results of the reliability testing programs for our review. The successful culmination of such a test program provides sufficient bases to conclude that the HPCS diesel generator has the reliability required by General Design Criterion 17. These documentation is expected soon and we will address our review of these results in a supplement to this report.

IEEE 387-1977 Section 5.6.2.2(1) and Regulatory Guide 1.108 position C.1.b.3 recommend that the periodic testing of diesel generator units should not impair the capability of the unit to supply emergency power within the required time. The diesel generator unit design should include an emergency override of the test mode to permit response to bona fide emergency signals and return control of the diesel generator unit to the automatic control system.

The override feature has not been provided in the WNP-2 diesel generator unit design, and therefore the system as designed is not acceptable. We will pursue this item with the applicant and provide our results in a supplement to this report.

Branch Technical Position ICSB 17 (PSB) (in Appendix 8A of the Standard Review Plan) requires that diesel generator protective trips be bypassed when the diesel generator is required for a design-basis event. All protective trips are allowed during periodic testing. The allowed exceptions to the above

PSB - ELECTRICAL

ITEM Agenda Item #2
(from 9/25/81 meeting)

REMARKS: The SER (p. 14) requires that "the diesel generator unit design should include an emergency override of the test mode to permit response to bonafide emergency signals and return control of the unit to the automatic control system." This requirement is fully met by diesel generators 1 and 2. In the case of diesel generator 3 the requirement is met except during the following condition:

> Off-site power is lost while the generator is in parallel with the off-site source and the emergency loads are running.

Justification: When off-site power is lost while the HPCS unit is in parallel with the off-site power source, the emergency bus in automatically isolated from that source. Assuming a maximum droop setting of 5% and a diesel generator load of 100% (supplied to the system) the frequency change after the off-site power is lost is given by $\Delta f(\%) = 5 \times (1 - \% \frac{\text{load}}{100})$ plus or minus the speed regulation. This does not appear to be critical considering that manual control of both speed and transfer to isochronous mode can still be done. Furthermore, the ADS system, supplied from the station batteries, provide a redundant backup to the HPCS system.

See also the attached Supply System to NRC letter, GO2-81-327, Oct. 2, 1981.

RESOLUTION: This response is satisfactory. This item is closed.

GD Bouchey	-396	Docket File
KD Cowan	-927M	Chrono File
LT Harrold	-410	kf/file
BA Holmberg	-904A	BAH/lb
JD Martin	-927M	KDC/lb
RG Matlock	-901A	RGM/lb
TL Meade	-927M	TLM/lb
JW Shannon	-660	sf (2)
GC Sorensen	-440	

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO. ____________

Docket No. 50-397

G02-81-327
October 2, 1981

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Schwencer:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2
HPCS DIESEL TESTING

The following is in response to Mr. Sang Rhow's concern about HPCS diesel testing expressed during the Power Systems Branch, Supply System Meeting September 25, 1981.

Previous switchgear logic prevented the parallel of the HPCS diesel and station service power. That logic has been modified and the HPCS diesel now can be paralleled to station service.

Regulatory Guide 1.108 dictates diesel testing requirements, one of which is full rated load testing. The total load of SM-4, including the HPCS pump is not enough to fulfill the full rated load test. Thus the capability to synchronize and parallel with the station service was necessary.

Periodically during plant operation the HPCS diesel will be manually started and loaded. It will be separately synchronized to the 230KV startup offsite power source and loaded. Functional testing of the automatic control circuitry is conducted on a periodic basis to demonstrate proper operation.

The actual test procedure has not been written at this time. It is our intent to comply with Regulatory Guide 1.108, concerning diesel testing requirements. Synchronizing and paralleling with station service is necessary to comply with these requirements.

FILE COPY

AUTHOR:	T. L. Meade *T.L. Meade*		FOR SIGNATURE OF:	G. D. Bouchey
SECTION				
FOR APPROVAL OF	KD Cowan	BA Holmberg	GC Sorensen	RG Matlock
APPROVED	*[signature]*	*[signature]*	*[signature]*	*[signature]*
DATE	10/1/81	10/1	10/1/81	

Mr. A. Schwencer
Page 2
G02-81-327
October 2, 1981

Administrative procedures will be used to prevent concurrent testing of more than one diesel.

Please contact us if further information is necessary.

Very truly yours,

G. D. Bouchey
Deputy Director
Safety & Security

GDB/TLM/ldm

cc: WS Chin - BPA
AD Toth - NRC Resident
NS Reynolds - Debevoise & Liberman
JC Plunkett - NUS Corporation
R Auluck - NRC DC
S Rhow - NRC DC
OK Earle - B&R RO
EF Beckett - NPI
WNP-2 Files

9.1.4

9.1.4 Fuel Handling System

The fuel handling system in conjunction with the fuel storage area provides the means of transporting, handling and storing of fuel. The fuel handling system consists of equipment necessary for the safe handling of the spent fuel cask and for safe disassembly, handling, and reassembly of the reactor vessel head and reactor internals during refueling operations. The system also includes additional equipment designed to facilitate the periodic refueling of the reactor.

The entire system is housed within the reactor building which is seismic Category I and flood protected. In addition, the reactor building provides tornado protection up to the refueling floor, elevation 607 feet. The jib cranes and the reactor building crane are seismic Category I. Other components of the fuel handling system are attached to the fuel pool wall or are to large to fall into the pool. [The applicant has not provided an acceptable response to our concern of lifting (dropping) an object lighter than a fuel assembly with the handling tool which could have a greater kinetic energy.] However, fuel handling system components are not required to function following an SSE. The 125-ton spent fuel cask handling crane is used for handling the 125-ton spent fuel shipping cask and is used to move the reactor vessel head, shroud head/ separator and dryer assembly. The refueling platform which travels over the spent fuel storage racks is designed to seismic Category I requirements from a structural standpoint. The design thus satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." [We will report on the lifting of light loads in a supplement to this SER.]

Q. 010.055
(9.1.2)

Describe, discuss, and verify that the maximum potential kinetic energy contained in all objects of less weight than a spent fuel assembly which will be handled over spent fuel will not exceed the effects of the fuel handling accident described in section 15.7.4 of the FSAR.

Response:

Table 010.055 has been prepared as a response to this question. This response also closes out an open item from the Auxiliary Systems Branch meeting October 7, 1981.

TABLE 010.055-1

LIGHT LOADS OVER THE SPENT FUEL POOL

Item	Distance Above Pool Surface or Above Fuel Rack in Pool --(in feet) Above Pool	Above Rack	Weight No.	Kinetic Energy at Impact (top of rack) ft/#
1. Channel Bolt Wrench	4		40	982
2. Channel Handling Tool	14		75	2,592
3. Channel Guaging Fixture		13*	210	2,389
4. General Purpose Grapple	14		25	864
5. Clam Shell Retriever	4		14	344
6. Manipulator Grapple	4		50	1,228
7. Actuating Pole	3#	20.5*	100	2,075
8. General Area Under-water Light	4		40	982
9. In Core Detector Cutter	4		150	3,684
10. Fuel Support Grapple	4		147	3,610
11. Peripheral Orifice Grapple	4		45	1,105
12. Peripheral Orifice Holder	4		130	3,193
13. Blade Guide	3*	20.5*	170	3,527
14. Fuel Bail Cleaner	4		100	2,456
15. Grid Guide	3#	20.5*	175	3,631

* Distance of CG to top of rack.
\# Distance of CG above pool.

TABLE 010.055-1

LIGHT LOADS OVER THE SPENT FUEL POOL

16.	Dummy Fuel Assembly		8.2	600	4,305
17.	Peripheral Fuel Support Plug	4		300	7,368
18.	Fuel Grapple		23.5	100	2,056
19.	Control Tube Grapple	4		45	1,105
20.	Guide Tube Grapple	4		35	860
21.	Control Rod Latch Tool	4		45	1,105
22.	Fuel Bundle Sampler	4		650	15,964
23.	Fuel Bundle & Channel		8.2	697	5,001
24.	Fuel Bundle & Channel w/Grapple[1]		8.2/15/7	697/100	6,375

* Distance of CG to top of rack.
# Distance of CG above pool.

Note 1: Assumed to be fuel grapple.

DSER 9.1.4

The spent fuel cask storage and loading pool is adjacent to the spent fuel pool, separated from the fuel pool by two feet thick reinforced concrete walls and is isolated from the pool by a gate. The travel path of the spent fuel cask will be controlled by means of a travel path and interlocks such that it will not be transported over any safety-related equipment or the spent fuel pool. Thus, the requirements of General Design Criterion 61, "Spent Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," are satisfied for handling of the spent fuel cask.

Generic Task A-36 has been resolved by NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" which was transmitted to the applicant for action by generic NRC letters dated December 22, 1980 and February 3, 1981. NUREG-0612 provides guidelines for necessary changes to assure safe handling of heavy loads once the plant becomes operational. Enclosure 2 attached to the December 22, 1980 generic letter identified a number of measures dealing with safe load paths, procedures, operator training and crane inspections, testing and maintenance. [The applicant has not responded to the generic letters nor committed to implement these interim actions prior to the final implementation of NUREG-0612. We will require that the applicant commit to implement these interim actions prior to the receipt of their operating license. We will report resolution of this item in a supplement to this SER.]

Open SER Issue

9.1.4 NUREG-0612

A response to this issue was submitted January 13, 1982 in letter G02-82-32.

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

January 13, 1982
G02-82-32
SS-L-02-CDT-82-012

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSE TO NUREG-0612
CONTROL OF HEAVY LOADS

Reference: Letter, D.C. Eisenhut to all Licensees, et al, "Control of Heavy Loads," dated December 22, 1980

Enclosed are sixty (60) copies of the WNP-2 response to NUREG-0612, "Control of Heavy Loads" transmitted via the reference letter. The WNP-2 draft SER open item on this subject should be closed upon receipt of this response.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca

Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

DSER 9.4.8

located in the seismic Category I, tornado protected diesel generator building. Thus the guidelines of Regulatory 1.29, "Seismic Design Classification," are met. [The applicant has not provided a description of the effects of losing the system as the result of tornado missiles entering through the louvers or loss of the air handling unit as a single failure concurrent with the tornadic event. We cannot provide any conclusions as to conformance with General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases" and the guidelines of Regulatory 1.117 relating to the protection against the effects of tornado missiles. We will report resolution of this item in a supplement to this SER].

The cable cooling system is not located near any high or moderate energy piping, thus, the guidelines of Branch Technical Position ASB 3-1 concerning high and moderate energy pipe breaks, and is, therefore, acceptable.

Based on the above, we conclude Diesel-Generator area cable cooling system is in conformance with the guidelines of Regulatory Guides 1.29 and Branch Technical Position ASB 3-1 relating to the system's seismic classification and high and moderate energy pipe breaks and is, therefore, acceptable. [We cannot conclude that the requirements of General Design Criteria 2 and 4 as relates to protection against natural phenomena, environment and missile effects and the guidelines of Regulatory Guide 1.117 concerning protection against the effects of tornado missiles are met until the applicant provides acceptable additional information. We will report resolution of this item in a supplement to this SER.]

Open SER Issue

9.4.8 <u>Diesel-Generator Area Cable Cooling System</u>

See revised FSAR page 9.4-50 attached.

A387

9.4.8 DIESEL-GENERATOR AREA CABLE COOLING SYSTEM

9.4.8.1 Design Bases

The critical electric cabling which runs between the emergency diesel-generators and the main control room and critical switchgear room is routed in corridors adjacent to the diesel-generator building and in corridors between the reactor building and radwaste building. These corridors are normally ventilated by the turbine building and radwaste building ventilation systems; however, an emergency cable cooling ventilation system is provided to ensure that ambient temperatures in the corridors do not exceed 115°F, the ambient environmental temperature for which the cable is rated, in the event of loss of offsite power. During an extreme winter outside temperature of -27°F, the incoming air is heated to a minimum temperature of 35°F.

The ventilation system is comprised of two independent and separate systems which cool Division 1 and Division 2 cable areas. Since these systems operate independently to cool their respective cables, a failure in one system will not effect the operational functions of the other cooling system. ~~The means of protecting system vents and louver from missiles is discussed in 3.5.~~ (SEE INSERT)

All components in the system are Seismic Category I, Quality Class I. The system fans are constructed and rated in accordance with AMCA standards.

9.4.8.2 System Description

The cable cooling system is shown in Figure 9.4-7. The system is composed of two exhaust fans powered from the Division 1 emergency power bus and one supply air handling unit powered from the Division 2 emergency power bus.

One of the two propeller type exhaust fans operates continuously to aid in normal ventilation of the corridors. The second exhaust fan, which is normally in standby, is started automatically when the Division 1 diesel-generator is started. The operation of this standby exhaust fan opens the outdoor air bypass damper when outdoor air temperature is above 40°F and when the Division 2 supply fan is not running. The operation of both Division 1 fans ensures necessary airflow through the corridors in which Division 1 cable is routed.

The Division 2 air handling unit is composed of a 30 kW electric blast coil heater, a water cooling coil supplied by standby service water system, and a centrifugal fan in a sheet metal housing. It is normally in standby. When the

Insert to page 9.4-50

The fresh air intake opening is located in the south exterior wall and is shielded by a concrete barrier which would preclude entry by a missile such as generated by a tornado. Division I exhaust fan discharges air into the HPCS diesel generator pipe chase. The chase opening to the atmosphere is through a concrete structure which is shielded by a labyrinth. Division II exhaust fan discharges air into a cable chase which is not open to the atmosphere. The air is removed by the radwaste building exhaust system.

9.4.12 Makeup Water Pump House Heating and Ventilating System

The makeup water pumphouse heating and ventilating system consists of two full capacity air handling units, each consisting of a fan, heater, cooling coil, and filter. One air handling unit normally operates with the second unit on standby. The makeup water pumps are not needed under a seismic event but are needed during a tornado (Refer to Section 9.2.5 of this SER). Therefore, the makeup water pump house HVAC is non-seismic Category I but is tornado protected. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena", are met. [The FSAR states the vents and louvers are missile protected by reference to Section 3.5. Section 3.5 of the FSAR does not specify that the vents and louvers of the makup water pump house HVAC are missile protected. Thus, we cannot conclude conformance with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Bases", are met. We will report resolution of this item in a supplement to this SER].

Based on the above, we conclude that the makeup water pump house heating and ventilation system is in conformance with the requirements of General Design Criterion 2 as related to protection against natural phenomena, are met. [We cannot conclude conformance with the requirements of General Design Criterion 4 as relating to protection from environmental and missile effect, and the guidelines of Regulatory Guide 1.117 as relating to protection against tornado missiles, as being met. We will report resolution of this item in a supplement to this SER.]

Open SER Issue

9.4.12 Makeup Water Pump House Heating & Ventilating System

See revised FSAR page 9.4-63 attached.

A 288 ASB 9.4.12
SER 9.4.12

the loss of offsite power, the system is powered from the emergency diesel generator buses. ~~The means of protecting the system vents and louvers from missiles is discussed in 3.5.~~ THE FRESH AIR INTAKE AND EXHAUST AIR OPENINGS ARE LOCATED IN THE EAST EXTERIOR WALL AND ARE SHIELDED BY A CONCRETE BARRIER WALL WHICH PRECLUDES ENTRY OF A MISSILE SUCH AS GENERATED BY TORNADO.

The make-up water pumps and auxiliaries are not required to operate in the event of a safe shutdown earthquake; therefore, all components of the heating and ventilating system serving the pump house are designed to Seismic Category II requirements as defined in 3.2. The system fans are constructed and rated in accordance with applicable AMCA standards.

9.4.12.2 System Description

The make-up water pump house heating and ventilating system is depicted in Figure 9.4-7. It consists of two full capacity air handling units and two battery hood exhaust fans which service the electric equipment area, and two full capacity fan coil units and two electric space heaters which service the pump area. Equipment details are given in Table 9.4-7.

The two air handling units serving the electric equipment area are each 8,000 cfm capacity units and consist of an insulated sheet metal cabinet housing a replaceable roughing filter, a 2 stage 40 kW electric blast coil heater, a water cooling coil, and a centrifugal fan. One of the two air handling units operates at all times to maintain design temperatures in the electric equipment area. The second unit is in standby and starts in the event that the operating unit fails.

The air handling units draw air from the outside atmosphere through intake louvers. The air is discharged, via ductwork, into the electric equipment area from which it flows into the pump room. It is then released either to the outside atmosphere, via relief dampers, or is partially recirculated back through the unit. Motor operated dampers on the unit intake ducts are so arranged that the unit can draw 100 percent outdoor air (8000 cfm) or recirculate a maximum of 6700 cfm of air drawn from the pump area back through the unit. The damper motor is controlled by a temperature switch which senses outdoor temperature. The damper will be positioned for 100 percent outdoor air when the outside temperature is between 50°F and 70°F.

The fan coil units servicing the pump area consist of a centrifugal fan, a water cooling coil, and a roughing filter in a sheet metal housing. The units recirculate air in the pump room only, and are interlocked electrically with the make-up water pumps to start when the pumps start. Each unit has sufficient capacity (16,500 cfm) to maintain design conditions

Evaluation and Findings

The applicant has committed to a post-accident sampling system that meets the requirements of NUREG-0737, Item II.B.3 in Amendment 17, but has not provided the technical information required by NUREG-0737 for our evaluation. Implementation of the requirement is not necessary prior to low power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage consistent with the conditions stated below.

1. Demonstrate compliance with all requirements of NUREG-0737, II.B.3 for sampling, chemical and radionuclide analysis capability, under accident conditions.
2. Provide sufficient shielding to meet the requirements of GDC-19, assuming Reg. Guide 1.3 source terms.
3. Commit to meet the sampling and analysis requirements of Reg. Guide 1.97, Rev. 2.
4. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

Open SER Issue

10.4.6 NUREG-0737/II.B.3 Post Accident Sampling

A response to this issue is provided in the TMI submittal (Appendix B).

DSER 11.4

11.4 flow in the vicinity of the compactor ~~baler~~ is exhausted by a fan through a HEPA filter to the radwaste building ventilation system to reduce the potential for airborne radioactive dusts. ~~We estimate the "dry" solid waste total to be [illegible] ft^3/year per reactor with a total activity content of <5 Ci. We find the design for the "dry" SWS acceptable.~~

"Wet" solid wastes consisting of concentrated chemical waste evaporator bottoms, chemical drain tank effluents and spent resin sludges will be treated by a volume reduction and solidification system at the Washington Nuclear Project, Unit No. 2. The solidification system will be a cement-silicate system, providing for solidification of the above wastes in steel containers. The solidifying agent will be a mixture of Portland cement and sodium silicate, with proportions being determined in accordance with a process control program; the process control program, however, has not as yet been submitted and, therefore, has not been reviewed. Prior to operation, the applicant will be required to submit a process control program to assure complete solidification of all "wet" waste in conformance with the guidelines of Standard Review Plan 11.4 of NUREG-0800(5)(6).

INSERT "A", "B"

11.5 Process and Effluent Radiological Monitoring And Sampling Systems

The process and effluent radiological monitoring systems are designed to provide information concerning radioactivity levels in

Open SER Issue

11.4 Process Control Program

See revised FSAR pages 11-vi, 11.4-11 and 11.4-11a through 11.4-11d.

TABLE OF CONTENTS (Continued)

Insert A

Insert A

11.4.3.2 Process Control Program

11.4.3.3 Process Control Systems

11.4.3.4 Process Control Interlocks

11.4.3.5 Process Control Logic

11.4.3.6 Setpoints and Operating Control

11.4.3.7 Laboratory Verification of Formulations

11.4.3.8 Preoperational Testing

11.4.3.9 Maintenance, Calibration, and Formulation Control

11.4.3.10 Unanticipated Wastes

radwaste control room. This level transmitter also drives a level indicator on the local control panel and provides control functions for the decant pump, the sludge discharge pump, and the phase separator inlet selector valve. Sludge level indication is accomplishd by a pair of ultrasonic probes positioned in the phase separator.

11.4.2.13.2 Condensate Phase Separator Instrumentation

The condensate phase separators level instrumentation is the same as that described for the reactor water cleanup phase separators.

11.4.2.13.3 Waste Sludge Phase Separator Instrumentation

The waste sludge phase separator has total liquid level indication. It uses an air bubbler and a pressure sensing level transmitter. In addition to the level gage and high level alarm in the radwaste control room, the level transmitter provides control inputs to the decant pump, the stop and flush circuit on the sludge discharge pump, and the discharge valves from the waste collector and floor drain collector tanks to the waste sludge phase separator.

11.4.2.14 Spent Resin Tank Instrumentation

Level indication for the spent resin tank is essentially the same as that described for the cleanup phase separators utilizing an air bubbler and level transmitter for total liquid level and ultrasonic probes for resin level.

11.4.2.15 Concentrated Waste Measuring Tank Instrumentation

This tank is equipped with a level transmitter that drives a level indicator in the radwaste control room.

11.4.2.16 Waste Mixing Tank Instrumentation

The waste mixing tanks are equipped with ultrasonic level detectors that drive indicators, a recorder and high level alarms on the solid waste control panel. They also provide control signals to stop the centrifuges on high level. The waste mixing tanks are also provided with radiation detectors. These monitors which have a range of 10 mR/hr to 100 R/hr drive a recorder and alarms on the solid waste control panel.

Insert B

Insert B

11.4.3 PROCESS CONTROL PROGRAM

11.4.3.1 Objective

The objective of the Process Control Program is to assure the complete solidification of all wet wastes. To meet this objective the Process Control Program has incorporated the recommendations set forth in NUREG 0800, Branch Technical Position - ETSB 11-3 and NUREG 0473.

11.4.3.2 Process Control Program

The cement-sodium silicate solidification process is designed to produce a freestanding solid with essentially no free liquid. Due to the latitude of waste and cement proportions that will solidify under the influence of sodium silicate, the solidification system can be operated with mixing ratios that assure solidification occurs even with nominal waste stream variations.

To assure that the system will produce an acceptable solidified product, the following process control elements have been incorporated:

- Process control systems
- Process control interlocks
- Process control logic
- Setpoints and operating limits control
- Laboratory verification of formulations
- Preoperational testing
- Maintenance, calibration, and formulation control
- Unanticipated wastes

11.4.3.3 Process Control Systems

The processing and material handling equipment is fully instrumented and the entire operation from mixing and filling to placing containers in storage is monitored and controlled from the solid waste system control panels.

The levels of waste and solidification materials are monitored at key points in the system using ultrasonic sensors. Pressure switches on the discharge of each proportioning pump give positive indication of pump operation and the flow of process materials. A flow sensor provides positive indication of cement flow. The waste mixing tank temperature is monitored and any temperature outside of preset limits is annunciated at the control panel. The level of waste-cement mixture, in the disposable container, is monitored with an ultrasonic sensor and the flow of waste-cement mixture is automatically stopped upon sensing high level by the ultrasonic signal.

The flow monitoring system provides a permanent record of the quantity of waste and solidification agents in each container by means of a four-pen recorder.

The solidification process is selected, initiated, and monitored at the solid waste system control panel. The control panel contains a graphic display with system control switches, indicators, and readouts arranged in mimic tracings for ease and accuracy of operation. The control panel graphic display includes valve position indicating lights, motor operation indicating lights, level indicators for storage tanks and the waste container under the fillport, alarm annunciators, process select and master control switches, closed-circuit television monitors, indication monitor readouts, and controls and indicators for the disposable container handling equipment.

11.4.3.4 Process Control Interlocks

Process control interlocks prevent system operation if components malfunction or inadvertent lineups are made. These interlocks ensure that the system operates to solidify wet waste only if the following conditions are met:

- A waste container is in place under the fillport.
- The fillport seal plate is down.
- The waste container is not full.
- The waste mixing tank and piping heat tracing is energized and above the minimum required temperature.
- The waste tank mixer is operating.
- Cement aeration blower and bag filter are operating.
- Waste tank level is above minimum.
- Cement storage tank and feed hopper levels are above minimum.
- Sodium silicate day tank level is above minimum.
- Cement is actually flowing.
- Sodium silicate is actually flowing.
- Waste is actually flowing.
- Waste-cement mixture is actually flowing.

The process selector switch and master start switch are interlocked such that the initial process would continue even if the process selector position were changed and/or the master start switch were depressed again.

The ultrasonic level monitors on the bulk storage tanks prevent system startup if either insufficient cement, waste, or sodium silicate is available for a complete process run.

11.4.3.5 Process Control Logic

The solidification system contains a logic control unit that controls the sequence and duration of process operations. The control unit contains logic control steps designated to perform internal checks of system conditions prior to initiation of subsequent process operations. Continuation of the process is dependent upon satisfying the conditional setpoints. Any time during a process run that a setpoint is exceeded the process is automatically stopped and annunciated. Conditional setpoints will be determined based on preoperational test results.

11.4.3.6 Setpoints and Operating Control

The laboratory verification of solidification formulations and the confirming data developed by full scale preoperational testing(s) will determine the setpoint values for each component in the system. Setpoints for parameters that are critical to the solidification process are preset to assure operation at the required conditions. The critical setpoint conditions are segregated in a locked subpanel in the rear of the main process control panel. This provides for direct administrative control of access to and adjustments of the control system setpoints.

The preset values for setpoints and the preset ratios of waste, cement, and sodium silicate are different for each particular type of waste. The proper values are automatically selected by an operator-controlled master switch with positions labelled according to the type of waste to be processed.

11.4.3.7 Laboratory Verification of Formulations

The design of the solidification system is based on laboratory, pilot plant, and full-scale system studies of each type of waste. Laboratory verification allows setpoint adjustments to compensate for plant variations from typical formulations.

Successful solidification of wet wastes is assured by development of solidification blends based on plant-waste composition coupled with laboratory solidification studies.

11.4.3.8 Preoperational Testing

The preoperational testing program will be designed to functionally test the solidification equipment under all modes of operation. The test program, in conjunction with the laboratory verification program, will determine the optimum setpoints and operating parameters necessary to insure the solidification of all wet waste matrices. In addition, the test program will also define the minimum and maximum parameter boundaries that still produce a freestanding solid with essentially no free liquid.

The solidified waste containers will be sectioned and the contents examined for homogeneity and the absence of free liquid. The results of this inspection will be documented and compared to the laboratory verification program results. If the results from both tests indicate the presence of a freestanding solid with essentially no free liquid, this information will then be recorded and used for future reference. Discrepancies between the laboratory verification program results and the actual results will be documented and resolved.

11.4.3.9 Maintenance, Calibration, and Formulation Control

Control of solidification parameters is assured by a maintenance, calibration, and formulation control program. The Operation and Maintenance Manual, supplied by the vendor, recommends specific maintenance and calibration frequencies for the various system components. The manual also recommends that a periodic verification of the quality and condition of the cement and sodium silicate in the plant storage tanks be performed. Laboratory verification of the effectiveness of the solidification formulations are performed to determine if system setpoint adjustments are required to maintain optimum product quality.

11.4.3.10 Unanticipated Wastes

From time-to-time it will become necessary to solidify wet wastes which have never been verified by laboratory formulation. Examples of such wastes are; decontamination solutions and laundry detergents.

When this occurs the solidification requirements must be determined on a case-by-case basis using the laboratory verification of formulation program. Records and results of such testing will be retained for future use.

DSER 12.3.4

12.3.4 <u>Area Radiation and Airborne Radioactivity Monitoring Instrumentation</u>

The objectives of the applicant's area radiation monitoring system are; 1) to warn of excessive gamma radiation levels in areas where nuclear fuel is stored or handled; 2) to provide operating personnel in the control room with a continuous record and indication of gamma radiation levels at various locations throughout the plant; 3) to assist in the detection of unauthorized or inadvertent movement of radioactive material; and 4) to warn of increased radiation levels by alarming when the radiation levels exceed present levels. In order to meet these objectives, the applicant plans to use thirty area monitors, located in areas where personnel may be present and where radiation levels could become significant. These monitors will be equipped with local and remote audible alarms and a facility for central recording. As a result of our review, all area monitors located in high noise areas will be equipped with visual alarms as well. Area monitors will have a range of four decades, except for the two containment high-range radiation monitors provided in response to NUREG-0737. These monitors will have a range of seven decades, from 10^{0} to 107 R/hr. The applicant has provided area monitors in fuel storage and handling areas in accordance with 10 CFR Part 70.24(a)(1) and Regulatory Guide 8.12, "Criticality Accident Alarm Systems". Regulatory Guide 1.97 (Revision 2) states that areas requiring post-accident access should have area monitors with a range of 10^{-1} R/hr to 10^{4} R/hr. WNP-2 has no monitors (except in the fuel area) with this upper range. This is an open item. All area radiation monitors will be calibrated once every 18 months.

Open SER Issue

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

See revised FSAR pages 12.3-23, 12.3.28, 12.3-29, and 12.3-30 (attached).

12.3.4.3 Specification for Area Radiation Monitors

The areas radiation-monitoring system is shown as a function block diagram in Figure 12.3-20. Each channel consists of a combined sensor and a converter unit, a combined indicator and trip unit, a shared power supply, and a shared multipoint recorder. All channels also have a local audible alarm auxiliary unit mounted near the sensor.

Each monitor has a upscale trip that indicates high radiation and a downscale trip that indicates instrument trouble. These trips sound alarms but cause no control action. The trip circuits are set so that a loss of power initiates an alarm.

The type of detector used is a Geiger-Mueller tube responsive to gross gamma radiation over an energy range of 80 KeV to 7 MeV. ~~Each detector range covers four decades~~. Detector ranges are given in Table 12.-1.

The overall accuracy within the manufacturer's design range of temmperature, humidity, line voltage annd line frequency variation is such that the actual reading relative to the true radinng, including susceptability and energy dependance (100 KeV to 3 MeV) is within 9.5% of equivalent linear full scale recorder output for any decade.

The calibrting frequency is once every 18 months and assures that drift does not exceed $\pm$ 0.2% of equivalent linear full scale recorder output for a 24-hour period or a $\pm$ 2% for a 30-day period.

Facilities for calibrating area radiation monitor units are provided for by means of a test fixture designed for use in the adjustment procedure for the area radiation monitor sensor and converter unit. It provides several gamma radiation levels between 10 and 250 mR/hr. The calibration unit source is cobalt-60. A cavity in the test fixture receives the monitor sensor. A window is located on the back wall of the cylindrical lower half of the cavity through which radiation emanates from the source to the sensor. A chart on each test fixture indicates the radiation levels available from the unit for the various control settings. For checking at higher radiation levels, a source of suffucient strength and energy levels is provided in a shielded test fixture.

TABLE 12.3-1

AREA MONITORS

Station No.	Location And Title	Range ~~(4 decades)~~
1	Reactor Bldg. Fuel Pool Area	10^2-10^6mR/hr
2	Reactor Bldg. Fuel Pool Area	1-10^4mR/hr
3	Reactor Bldg. New Fuel Area	10^2-10^6mR/hr
4	Reactor Bldg. Control Rod Hyd Equipment Area E	1-10^4mR/hr
5	Reactor Bldg. Control Rod Hyd Equipment Area W	1-10^4mR/hr
6	Reactor Bldg. S 589' Level	1-10^4mR/hr
7	Reactor Bldg. Neutron Mon. Sys. Drive Mech. Area	1-10^4mR/hr
8	Reactor Bldg. STGS Filters Area	1-10^4mR/hr
9	Reactor Bldg. Northwest RHR Pump Room	1-10^4mR/hr
10	Reactor Bldg. Southwest RHR Pump Room	1-10^4mR/hr
11	Reactor Bldg. Northeast RHR Pump Room	1-10^4mR/hr

TABLE 12.3-1 (Continued) Page 2 of 3

Station No.	Location And Title	Range ~~(4 decades)~~
12	Reactor Bldg. RCIC Pump Room	1-10^4 mR/hr
13	Reactor Bldg. HPCS Pump Room	1-10^4 mR/hr
14	Reactor ~~Turbine~~ Bldg. ~~Turbine~~ 471 Level	10^{-2} ~~1~~-10^4 ~~m~~R/hr
15	~~Front Standard~~ Reactor Bldg 501 Level	10^{-2}-10^4 R/hr
~~15~~ 16	Reactor ~~Turbine~~ Bldg. ~~Entrance~~ 606 Level	10^{-2} ~~1~~-10^4 ~~m~~R/hr
~~16~~ 17	Turbine Bldg. ~~Reactor Feed~~ Turbine ~~Pump Area 1A~~ Front Standard	1-10^4 mR/hr
~~17~~ 18	Turbine Bldg. ~~Reactor Feed~~ Entrance ~~Pump Area 1B~~	1-10^4 mR/hr
~~18~~ 19	Turbine Bldg. ~~Condensate~~ Reactor Feed ~~Pump Area~~ Pump Area 1A	1-10^4 mR/hr
~~19~~ 20	~~Main Control Room~~ Turbine Bldg. Reactor Feed Pump Area 1B	1-10^4 mR/hr
~~20~~ 21	~~Radwaste Bldg. Valve Room E~~ Turbine Bldg. Condensate Pump Area	1-10^4 mR/hr
~~21~~ 22	~~Radwaste Bldg. Valve Room W~~ Main Control Room	1-10^4 mR/hr
~~22~~ 23	Radwaste Bldg. ~~Sample Area~~ Valve Room E	1-10^4 mR/hr
~~23~~ 24	~~Reactor Bldg. North CRD Pump Area~~ Radwaste Bldg Valve Room W	1-10^4 mR/hr
25	Radwaste Bldg. Sample Area	1-10^4 mR/hr
26	Reactor Bldg. North CRD Pump Area	1-10^4 mR/hr

TABLE 12.3-1 (Continued) Page 3 of 3

Station No.	Location And Title	Range ~~(4 decades)~~
~~24~~ 27	Reactor Bldg. North 478' Level	$1\text{-}10^4$ mR/hr
~~25~~ 28	Radwaste Bldg. Hot Machine Shop	$1\text{-}10^4$ mR/hr
~~26~~ 29	Radwaste Bldg. Contaminated Tool Room	$1\text{-}10^4$ mR/hr
~~27~~ 30	Radwaste Bldg. Waste Surge Tank Area	$1\text{-}10^4$ mR/hr
~~28~~ 31	Radwaste Bldg. Tank Corridor Area North	$1\text{-}10^4$ mR/hr
~~29~~ 32	Radwaste Bldg. Tank Corridor Area South	$1\text{-}10^4$ mR/hr
~~30~~ 33	Radwaste Bldg. Radwaste Control Room	$1\text{-}10^4$ mR/hr

Note: Alarm settings for all of the above monitors will be selected to provide indication of any abnormal increase in radiation levels while minimizing false alarms.

~~25~~	~~Radwaste Bldg. Sample Area~~	~~$1\text{-}10^4$ mR/hr~~
~~26~~	~~Reactor Bldg. North CRD Pump Area~~	~~$1\text{-}10^4$ mR/hr~~

DSER 12.5.1

the authority for direct contact with the Plant Manager in matters of health and safety that could affect onsite and/or offsite personnel. The Health Physics/Chemistry Manager can interact directly with the Plant Manager during the meetings of the Plant Operations Committee. He reports at the same level as the Operations Supervisor. The items discussed above are in agreement with the criteria of NUREG-0731 and Regulatory Guide 8.8 (Rev. 3) and are acceptable.

Based on information transmitted to the staff via a phone call with the applicant, the Health Physics/Chemistry Manager at WNP-2 meets the qualification criteria of Regulatory Guide 1.8 for Radiation Protection Manager. The draft ANS 3.1 recommends that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, and 2 years experience in radiation protection, 1 year of which should be nuclear plant experience, 6 months of which should be onsite. The Health Physics Supervisor, who will serve as the backup to the RPM in his absence, satisfies these requirements, according to information transmitted by a phone conversation with the applicant. The staff must formally receive this information in writing from the applicant before we can resolve the issue of qualifications. This is an open issue.

NUREG-0731 and Section 4.5.2 of ANSI 18.1 specify that technicians have two years experience in their specialty. The applicant has proposed that technicians at WNP-2 function in two specialities, health physics and chemistry, with only two years of training. Health Physics Appraisals at operating plants have found that such combined health

physics/chemistry technicians have lead to poor performance in both specialities because the technicians do not receive adequate training, qualification and retraining in both specialities. Until the staff can discuss this item further with the applicant, we will consider this an open item.

The equipment, instrumentation, and facilities used to implement the radiological safety program at WNP-2 follow the guidance of the following regulatory guides where applicable:

- Regulatory Guide 8.3, "Film Bade Performance Criteria".
- Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters" (except for Section C.2.b on calibration/ response tests).
- Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable". (Rev. 3)
- Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions For a Bioassay Program".

WNP-2 has two main locker-change rooms. Temporary change areas, as well as personnel and equipment monitoring stations, are set up as necessary to control the spread of contamination. There are three facilities for equipment and tool decontamination at WNP-2. Other

Open SER Issue

12.5.1 Education of HP Supervisor/Chem Tech.

A response to this issue was submitted January 11, 1982, in letter G02-82-25.

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

January 11, 1982
G02-82-25
SS-L-02-CDT-82-007

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
EDUCATION AND QUALIFICATIONS OF
HEALTH PHYSICS SUPERVISOR AND TECHNICIANS

In response to open items on pages 16 and 17 of section 12.5.1 of the draft SER for WNP-2; enclosed are sixty (60) copies of the subject information. This information should close out these open items.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/ct
Enclosures

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

DSER 12.5.1

health physics facilities include a hot machine and hot instrument shop for work on contaminated equipment, a lab complex, and calibration facilities.

Health physics equipment used for radiation protection purposes includes; protective clothing and accessories, respiratory protection equipment, air sampling equipment, and emergency kits. Laboratory equipment includes a multichannel pulse height analyzer with associated GeLi and NaI detectors, beta-gamma proportional counters, and a liquid scintillation counting system for tritium determination. This equipment was chosen because of its capability to identify and measure all the nuclides encountered in a power reactor. Radiation and contamination survey instrumentation at WNP-2 includes various alpha, beta, gamma, and neutron survey meters. This equipment was selected to provide a wide range of monitoring requirements from picocurie quantity measurements for lab work to thousand R/hr ranges to be used in the event of emergencies. Regulatory Guide 1.97 states that portable instrumentation should be available to analyze and assess post accident conditions. The range should be 10^{-3} R/hr to $10^{\overset{4}{\cancel{5}}}$ R/hr for photons and 10^{-3} rads/hr to 10^{4} rads/hr for beta radiation and low-energy photons. Since WNP-2 has no such instrumentation, this is an open item.

The area and airborne radioactivity monitoring systems at WNP-2 were chosen to provide continuous surveillance of radiation levels within the plant and radioactive airborne effluents released from the plant. These systems are backed up by portable area and airborne monitors for use when localized monitoring is needed.

WNP-2

Open SER Issue

12.5.1 Post Accident Portable Monitor Range

See revised FSAR page 12.5-3 (attached).

b. Regulatory Guide 8.4 is implemented except for C.2.b, which states, "The calibration/response test result should not exceed ±10% of an exposure from a source traceable to the National Bureau of Standards". This is accepted on the minus side, but is considered excessively stringent on the positive side. Since the error on the positive side results in exposure conservatism to the worker, +20% is a more reasonable limit for rejection of a pencil dosimeter.

12.5.2.1 Criteria for Selection

a. Radiation and Contamination Survey Instrumentation - This equipment was selected to cover the wide range requirements extending from picocurie quantity measurements in the laboratory Ten ~~to the~~ thousand R/hour ranges necessary for emergency dose rate determinations. The laboratory instrumentation was chosen to provide capability for the quantitative and qualitative analyses required to identify and measure the radionuclides encountered in a power reactor. The portable instrumentation includes low level detection capabilities for alpha, beta, and gamma contamination and wide ranges of dose rate measuring instruments for beta, gamma, and neutron radiation. The criteria for quantity selection were to provide adequate available counting time for anticipated demand in the laboratory and sufficient portable instruments to cover normal operational and emergency requirements in all areas of the WNP-2 facility.

b. Airborne Radioactivity Monitoring - Basic criteria for selection of this equipment was to provide a means for determining radioactive airborne effluents released from the plant, and to effectively monitor airborne radioactivity levels within the plant environs. Provisions have been made for continuing response monitoring of noble gases discharged from gaseous release points from the reactor, radwaste and turbine building, and for continuous sampling of radioiodines and particulates at these same locations. Internal plant air monitoring instrumentation is used within these buildings with readout locally and in the

contained in Regulatory Guide 1.33, Rev. 2, March 1978 regarding the minimum procedural requirements for safety-related operations; (4) compliance with the guidance contained in ANSI 18.7-1976/ANS 3.2; and (5) the applicant's program for compliance with Task Action Plan (NUREG-0660) Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents", for the development of Emergency Operating Procedure Guidelines. Additionally, the applicant's program for compliance with Item I.C.1 of NUREG-0737 for the development of Emergency Operating Procedures will be reviewed and reported in a supplement to this Safety Evaluation Report.

B. Operating and Maintenance Procedure Program

The applicant has committed in FSAR Chapter 17, Quality Assurance, to a program in which all activities are to be conducted in accordance with detailed written and approved procedures meeting the requirements of Regulatory Guide 1.33, Rev. 2, March 1978, "Quality Assurance Program Requirements (Operation)", and ANSI 18.7-1976/ANS 3.2. However, FSAR Section 13.5 still refers to Rev. 1, June 1977 of Regulatory Guide 1.33. We require the applicant to modify Section 13.5 to refer to Revision 2 of the guide.

The applicant uses the following categories of procedures for those operations performed by licensed operators in the control room:

System Operations (Including Radioactive Waste Systems)

General Operation

Abnormal Conditions (Including Annunciator Response)

Emergencies

Surveillance

Open SER Issue

13.5.2 <u>Operating and Maintenance Procedure Program</u>

See revised FSAR page C.3.28 (attached).

Regulatory Guide 1.33, Rev. ~~1~~ 2, ~~June~~ Feb 8 1977 ~~7~~ 8

Quality Assurance Program Requirements (Operation)

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

Compliance or Alternate Approach Assessment:

Compliance is discussed in the topical report referenced in 17.2

Specific Evaluation Reference:

Refer to 13.5.1.1 and 17.2.

DSER 15

WNP-2 OPEN ITEMS

1. Overpressurization Protection (5.2.2)- The applicant must submit for our reivew and approval, a plant specific overpressurization analysis using the ODYN code and including the effect of recirculation pump trip.

2. Safety/Relief Valve Surveillance (5.2.2)- The applicant must commit to participate in a surveillance program to monitor the performance of safety/relief valves.

3. Pressure Interlocks on ECC Injection Valves (6.3)- The applicant must verify that interlocks are present at all times for both manual and automatic valve operation and that the interlocks do not allow valve opening until the reactor coolant pressure is below the design pressure of the ECC system involved, or provide an alternative configuration which satifies the requirements of SRP Section 6.3.

4. Premature LPCI Diversion (6.3)- The applicant must provide assurance that LPCI flow will not be diverted to containment cooling before adequate core cooling is provided. (We have accepted a discussion of emergency procedures and operator training for this item on other applications.)

5. Long Term Air Supply to ADS Valves (6.3)- The applicant must verify that the bottled air supply serving as a backup to the normal air supply for the ADS valves is valved in during normal operation, or provide justification as to why credit should be given to this air supply.

6. Thermal Power Monitor in Transient Analyses (15)- We require that the thermal power monitor time constant be included in the plant technical specifications or that no credit be taken for the thermal power monitor in transient analyses.

15. Thermal Power Monitor in Transient Analyses

The Thermal Power Monitor (TPM) function of the neutron monitoring system is addressed in the following WNP-2 Plant Technical Specifications:

T.S. Section	Title
2.2	Limiting Safety System Settings (Table 2.2.1-1)
3/4.2.2	APRM Setpoints
3/4.3.1	Reactor Protection System Instrumentation (Tables 3.3.1-1, 3.3.1-2, & 4.3.1.1-1)

The surveillance requirements on the thermal power monitor in Table 4.3.2.2-1 have been modified to require a measurement of the TPM time constant on a refueling outage frequency.

DSER 15

7. ODYN Reanalyses (15) - For thermal limit evaluation we require a re-analysis of BWR pressurization transients using the ODYN code.

8. Reclassification of Transients (15)- We require that the turbine trip without bypass and the generator load rejection without bypass events be classified as moderate frequency events and they satisfy the MCPR limit of 1.06.

9. Modification of ADS Logic (II.K.3.18)- We require the applicant to provide one of the following: 1) Analyses of containment heatup rates which demonstrate that a high drywell pressure signal will be present at a time early enough to preclude exceeding the criteria of 10 CFR 50.46 for a stuck open relief valve event or an outside steam line break, 2) a commitment to modify the current logic to either bypass the high drywell signal or add a timer to which initiates when Level 1 water level is reached and which bypasses the high drywell signal upon timing out. If the timer option is selected, an analysis supporting the time setting must be provided.

10. Loss of Power to Pump Seal Coolers II.K.3.25- We require verification by the applicant of the applicability of the BWR Owners Group Test Data to WNP-2.

11. Restart of Core Spray Systems (II.K.3.21)- We require that modifications be made to the HPCS system logic so that HPCS will automatically restart on a low water level signal after it has been manually terminated from the control room.

Open SER Issue
15. Reclassification of Transients

Refer to ODYN reanalysis (SER Section 15, RSB.7) for a response to this issue.

Open SER Issue

15 <u>ODYN Reanalyses</u>

A response to this issue was submitted January 11, 1982 in letter G02-82-26.

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

January 11, 1982
G02-82-26
SS-L-02-CDT-82-008

Docket No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSES TO REACTOR SYSTEMS BRANCH QUESTIONS
AND ODYN ANALYSIS WNP-2 FSAR REWRITE

Reference: Letter, R. L. Tedesco to R. L. Ferguson, "WNP-2 FSAR - Request for Additional Information", dated June 8, 1981.

Enclosed are sixty (60) copies of responses to the remaining Reactor Systems Branch questions. The responses to Questions 211.129 and 211.136 are new. The responses to 211.031 and 211.148 are rewrites of responses previously submitted to the NRC. The response to 211.209 was also submitted as a part of the LRG appendix (RSB-3). These responses will be incorporated into the FSAR in Amendment 23.

Also, enclosed are sixty (60) copies of the draft revised FSAR pages as a result of the ODYN analysis. These revised pages will also be incorporated into Amendment 23 to the WNP-2 FSAR.

Very truly yours,

G. D. Bouchey
Deputy Director, Safety and Security

CDT/ct
Enclosures

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

DSER 15.8

The GE Owners' Group is currently developing a set of Reactivity Control Guidelines, which will incorporate the above steps for mitigating ATWS events. The applicant's procedure for mitigating ATWS will be reviewed under the emergency operating procedure program as described in Section 13.5.2. The results of the staff review will be reported in a supplement to this Safety Evaluation Report.

Open SER Issue

15.8 ATWS Emergency Operating Procedures

The BWR Owners' Group, of which the Supply System is a member, is currently developing guidelines for preparation of emergency operating procedures including a guideline for ATWS. WNP-2 will prepare an emergency procedure for ATWS using the Owners' Group guideline.

The procedure will be sufficiently simple to permit prompt operator recognition of an ATWS and performance of mitigating actions. Direction will be provided for the interval from identification of the ATWS until all rods are inserted to more than position 06.

Prior to approval, the ATWS procedure will be verified correct by performing independent reviews and Control Room walk-throughs. Following approval, each operating shift will receive walk-through training. Operations personnel will be trained on the simulator when it becomes available; retraining will be conducted semi-annually.

The ATWS and other emergency procedures will be located in the Control Room and readily available to operating personnel. Procedure control measures will insure all operators are advised regarding changes.

DSER II.K.3.18, 25, 21.

7. ODYN Reanalyses (15) - For thermal limit evaluation we require a re-analysis of BWR pressurization transients using the ODYN code.

8. Reclassification of Transients (15)- We require that the turbine trip without bypass and the generator load rejection without bypass events be classified as moderate frequency events and they satisfy the MCPR limit of 1.06.

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10. Loss of Power to Pump Seal Coolers II.K.3.25- We require verification by the applicant of the applicability of the BWR Owners Group Test Data to WNP-2.

11. Restart of Core Spray Systems (II.K.3.21)- We require that modifications be made to the HPCS system logic so that HPCS will automatically restart on a low water level signal after it has been manually terminated from the control room.

WNP-2

Open SER Issues

II.K.3.18 Modification of ADS Logic (RSB-9)

II.K.21 Restart of Core Spray Systems (RSB-11)

II.K.3.25 Loss of Power to Pump Seal Coolers (RSB-10)

Responses to the above SER issues are included in the TMI submittal (Appendix B).

Open SER Issues

Attached as formal submittals are the responses to Containment Systems Branch issues CSB-3, 5, 10, 13, 14, 17, 27, 28, 34, 35, 36 and 42 which were closed at the 9/14/81 - 9/17/81 branch meeting.

Issues CSB-1, 6, 7, 8, 21, 22, 41 and 43 through 48 were addressed in the responses to draft SER issues.

CONTAINMENT SYSTEMS BRANCH

ISSUE 3

NRC:	Add statement that all motor-operated valves fail in the safe position in all conditions. Refers to 6.2-56.
Supply System	The resolution to this concern is as follows: All motor-operated valves fail as-is, however, check valves or redundant valves from alternate power sources are provided.
Summation:	This response is satisfactory.

ISSUE 5

NRC:	Indicate that the operator will be required to determine whether or not to close the feedwater block valves twenty minutes after indication of a large scale LOCA.
Supply System:	The Supply System has revised Section 6.2* (page 6.2-58) in Amendment 19 to the WNP-2 FSAR. This revision responds to NRC Question 022.074.
Summation:	This issue is closed.

* See attached FSAR page change.

FSAR Revision

022.074

Table 6.2-16 contains those influent pipes that comprise the reactor coolant pressure boundary and penetrate the containment.

6.2.4.3.2.1.1.1 Feedwater Lines

The feedwater lines are part of the reactor coolant pressure boundary as they penetrate the drywell to connect with the reactor pressure vessel. The isolation valve inside the drywell is a y-pattern check valve, located as close as practicable to the containment wall. Outside the containment is another y-pattern check valve located as close as practicable to the containment wall and farther away from the containment is a motor operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. However, in case a loss-of-coolant accident occurs without a seismic event, the design allows the condensate and condensate booster pumps to supply feedwater to the vessel through a bypass line around the reactor feed pumps - which are tripped on a loss of steam supply - as soon as the vessel is partially depressurized. For this reason, the outermost, gate valve does not automatically isolate upon signal from the protection system. The gate valve meets the same environmental and seismic qualifications as the outboard isolation valve. The valve is capable of being remotely closed from the control room to provide long-term leakage protection upon operator judgement that feedwater makeup is unavailable or unnecessary. INSERT No credit is taken for feedwater flow in accessing core and containment response to a loss-of-coolant accident.

6.2.4.3.2.1.1.2 HPCS Line

The HPCS line penetrates the drywell to inject directly into the reactor pressure vessel. Isolation is provided by an air testable check valve, located inside the drywell with position indicated in the main control room, and remote-manually actuated gate valve located as close as practicable to the exterior wall of the containment. Long-term leakage control is maintained by this gate valve. If a loss-of-coolant accident occurred, this gate valve would receive an automatic signal to open.

6.2.4.3.2.1.1.3 LPCI and LPCS Lines

Satisfaction of isolation criteria for the LPCI and the LPCS system is accomplished by use of remote-manually operated gate valves and check valves. Both types of valves are normally closed with the gate valves receiving an automatic

Question 022.074

Insert to 6.2-58

The operator can determine if make-up from the feedwater system is unavailable by use of the feedwater flow indicator in the control room, which will show high flow for a feedwater pipe break or no flow for feedwater pump trip.

The operator can also determine if make-up from the feedwater system is unnecessary by verifying that the ECCS is functioning properly and the reactor water level is being adequately maintained. ECCS operation signals and reactor vessel water level indication are provided in the control room for operator information.

Since due to the check valves it is not necessary to immediately isolate the feedwater system for leakage mitigation, there is no need to alert the operator to initiate the feedwater isolation signal other than as described above. However, for long-term isolation purposes, the operator may close the motor-operated gate valves at any convenient time.

Emergency procedures will require the operator to evaluate whether the operator should close the reactor feedwater block valves within twenty minutes following indication of a LOCA. If information indicates a degraded core condition prior to twenty minutes, the operator will take action to close the reactor feedwater block valve at that time.

CONTAINMENT SYSTEMS BRANCH

ISSUE 10

NRC: On page 6.2-141, WNP-2 agreed that the TIP System was GDC-55 and 56, originally 54. However, it can remain as 54.

Supply System: The Supply System has revised the response to Question 022.073 to show the criteria for the TIP System remaining GDC-54*.

Summation: This issue is closed.

* See attached page change.

Q. 022.073

In Table 6.2-16 of the FSAR, you indicate that the reactor recirculation hydraulic lines (X-76 and X-77) conform to the requirement of Criterion 57 of the GDC. It is our position that the isolation provisions for these specific lines should meet the requirements of Criterion 56. Further, in Table 6.2-16 of the FSAR, you indicate that traversing incore probe (TIP) system conforms to the requirements of Criterion 54 of the GDC. (Refer to Note 29 of Table 6.2-16.) However, in Section 6.2.4.3.2.3 of Criterion 57 of the GDC. It is our position that the TIP system should meet the requirements of GDC 56. Accordingly, revise Table 6.2-16 and other appropriate portions of the FSAR to reflect our position. Indicate if the other acceptable alternatives for meeting the requirements of the GDC as noted in Section 6.2.4 of the SRP could be applied to any of these lines.

Response:

During the Containment Systems Branch meeting September 14 - 17, 1981, the NRC agreed that the isolation provisions for these lines could meet the requirements of Criterion 54.

CONTAINMENT SYSTEMS BRANCH

ISSUE 13

NRC:	Page 6.2-123, RCIC steam supply indicates RCIC-V-8 valve open during normal operation, is this correct?
Supply System:	Yes, the valve is left open so that cold pipe is not thermal shocked. This mode of operation is a standard design.
Summation:	No action required.

CONTAINMENT SYSTEMS BRANCH

ISSUE 14

NRC:	Inerting will not be required until after commercial operation as defined by sustained 95% power operation.
Supply System:	The Supply System verifies WNP-2 will inert during commercial operation.

CONTAINMENT SYSTEMS BRANCH

ISSUE 17

NRC: When will containment inerting design be available?

Supply System: WNP-2 provided containment inerting description in docket letter committing to inerting, and is acceptable.

CONTAINMENT SYSTEMS BRANCH

ISSUE 27

NRC:	A $A/\sqrt{k}$ of 0.028 is used in steam bypass analysis. This is not acceptable. See Question 022.069.
Supply System:	Provided response to Question 022.069 using $A/\sqrt{k}$ = 0.050.
Summation:	F. Eltawila will review response and inform the Supply System of any further clarification required.

CONTAINMENT SYSTEMS BRANCH

ISSUE 28

NRC: Confirm that analysis for Question 022.069 does not take credit for heat sinks in the wetwell.

Supply System: The NRC will respond to the Supply System if more information is needed on this issue. Position is acceptable.

CONTAINMENT SYSTEMS BRANCH

ISSUE 34

NRC:	The Supply System downcomer vacuum breakers are flange mounted. Perform a Type B test on the flange after any replacement or removal of the vacuum breakers.
Supply System:	The Supply System proposes to perform a drywell/wetwell leak test at 5 psig to verify the pressure boundary, or perform an equivalent local test on the flange.
Summation:	This is an acceptable position.

ISSUE 35

NRC:	RHR heat exchanger thermal relief valve and Hx vent pipe length justification must be provided.
Supply System:	The problem is that the relief valve is a containment boundary but is located approximately 150 feet from the containment wall. The vent line shares a common discharge line with the relief valve and its isolation valve is also at the RHR heat exchanger so as not to negate the function of the relief valve. The NRC recognizes that the utility position is that the relief valve is located "as close as possible" to the containment wall.
Summation:	The Supply System will await NRC clarification of this generic concern.

CONTAINMENT SYSTEMS BRANCH

ISSUE 36

NRC:	WNP-2 ISI plan submitted to the NRC for review, does not provide for inspection of penetration weld on the SRVDL penetration through the downcomer. CSB will defer this issue to the MEB for further review.
Supply System:	The Supply System has performed an ASME Section III fatigue analysis per class 1 rules on this SRVDL penetration to verify that the usage factor is less than 1.0.
Summation:	The Supply System will await any further requests from the MEB.

CONTAINMENT SYSTEMS BRANCH

ISSUE 42

NRC:	NRC would like a comparison of quencher and arm tie-down load specifications versus Caorso test results.
Supply System:	The NRC will review NUREG-0487 and let the Supply System know whether more information is required.
	Analysis of quenchers for quencher arm loads and tie-down loads will be documented in the DAR and compared against Caorso test results.

WNP-2

Open SER Issues

Attached are formal responses to Containment Systems Branch issues CSB- 4, 9, 11, 12, 15, 16, 18, 19/24, 20, 23, 25/29, 26, 30, 31, 32, 33, 37, 38, 39 and 40, which required further documentation following the 9/14/81 - 9/17/81 branch meeting.

CONTAINMENT SYSTEMS BRANCH

ISSUE 4

NRC:	Make a general statement that test connections are tested.
Supply System:	The Supply System will add Note 7 to Figure 6.2-31a indicating that valves on test connections are also tested for containment isolation (App. J Type C test).
	See revised FSAR Figure 6.2-31a (attached).

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.

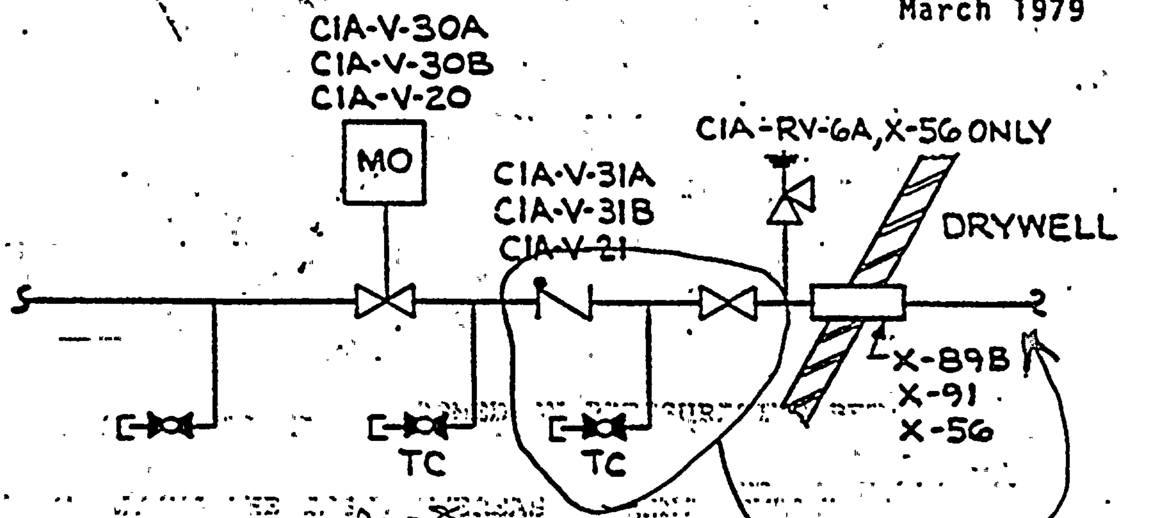
~~7. [illegible] Testing Is Per[illegible] By Pr[illegible] Between T[illegible] The One Inch Globe Valve Will Have [illegible] For [illegible] Isolation [illegible] And Under The Seat [illegible] Valve. The Difference Between Testing Under And Over The [illegible] 1 Inch Globe Valve Is Considered Negligible.~~

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

8. Type C Testing Will Be Performed On ~~All~~ Test Connections ~~Valves~~ That Are Located Between Containment Isolation Valves And Considered Part Of The Containment Isolation System.

RWH

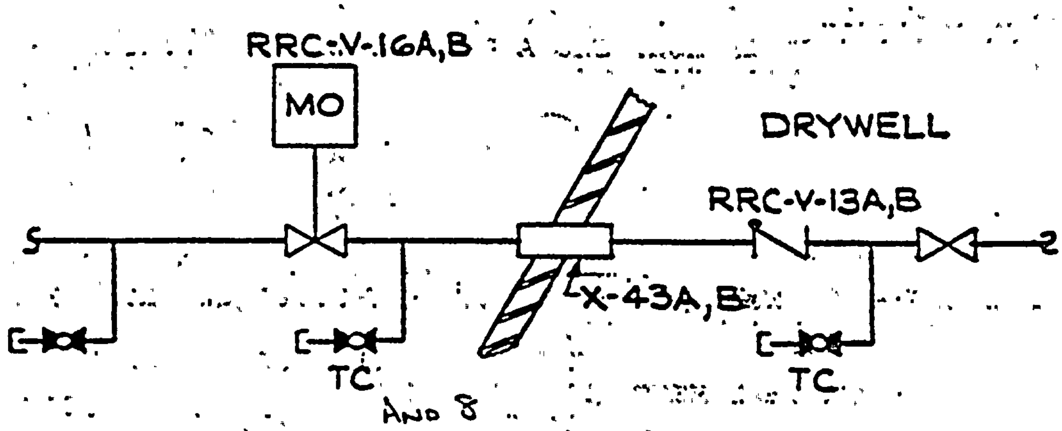
7. Type C testing is performed by pressurizing between the isolation valves. The 1" globe valve will have test pressure applied over the seat for the inboard isolation valve and under the seat for the outboard isolation valve. The difference between testing under and over the seat for a 1" globe valve is considered negligible.



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

See SCN 81-578

CONTAINMENT INSTRUMENT AIR

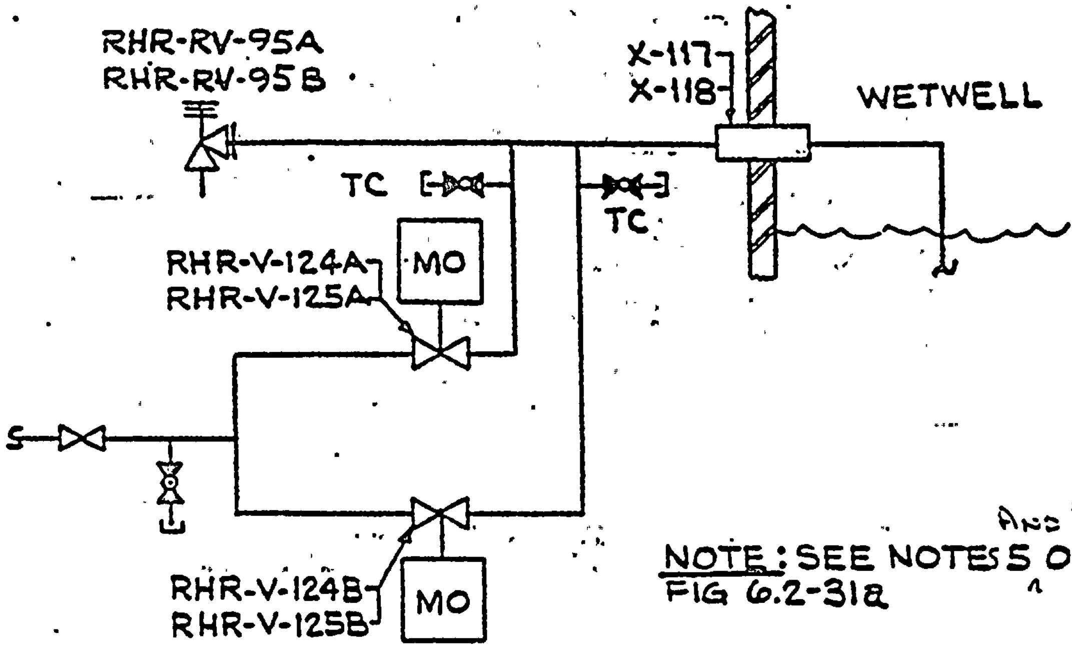


NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

RRC PUMP SEAL PURGE

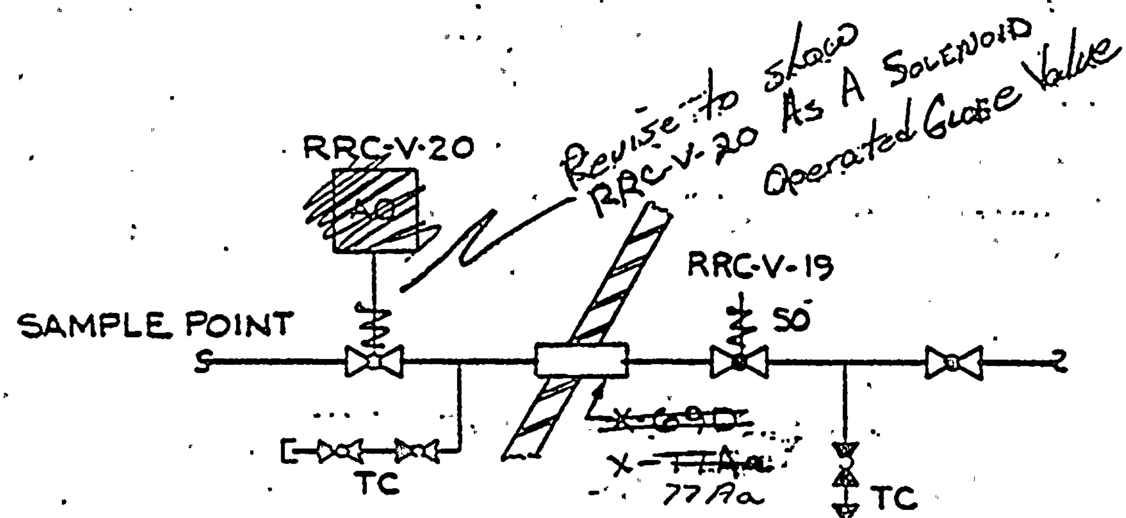
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-89B, X-91, X-56 & X-43A, X-43B	FIGURE 6.2-31c

Amendment No. 3
March 1979



NOTE: SEE NOTES 5 AND 8 ON FIG 6.2-31a

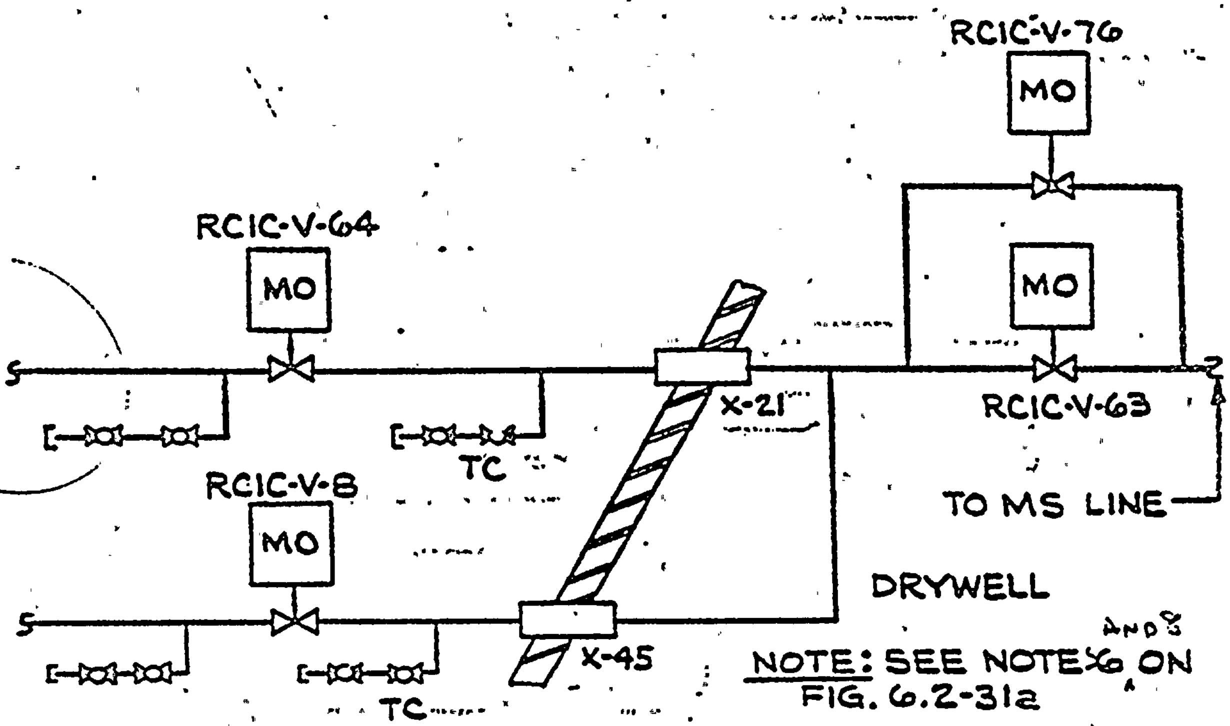
RHR STEAM RELIEF LINES



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

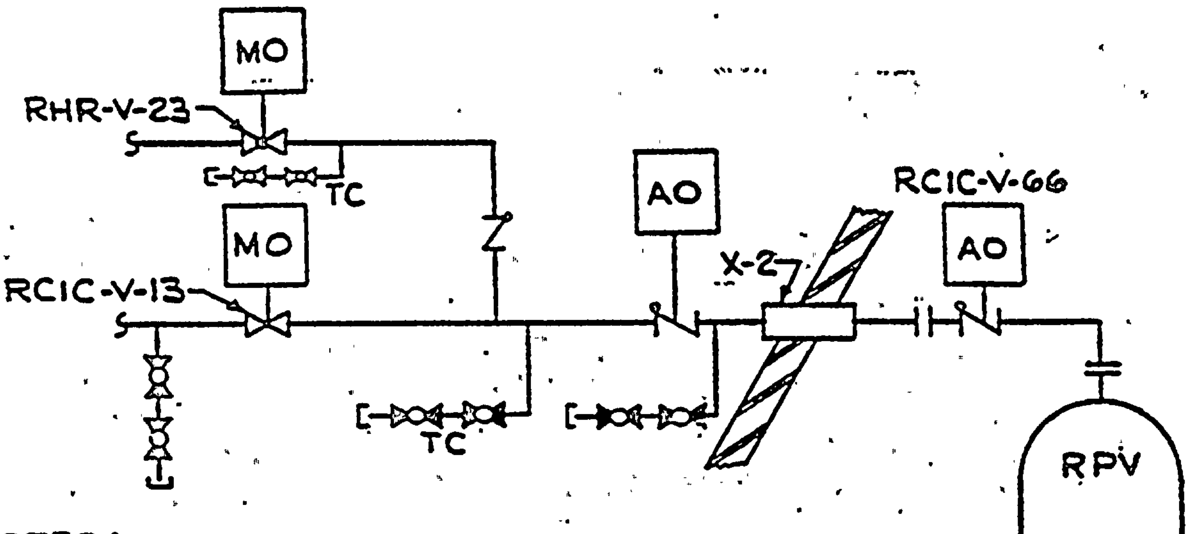
RRC SAMPLE LINE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-117, X-118 & X-~~69D~~ 77Aa	FIGURE 6.2-31d



NOTE: SEE NOTE 6 AND 8 ON FIG. 6.2-31a

STEAM TO RCIC TURBINE & RHR HEAT EXCHANGER

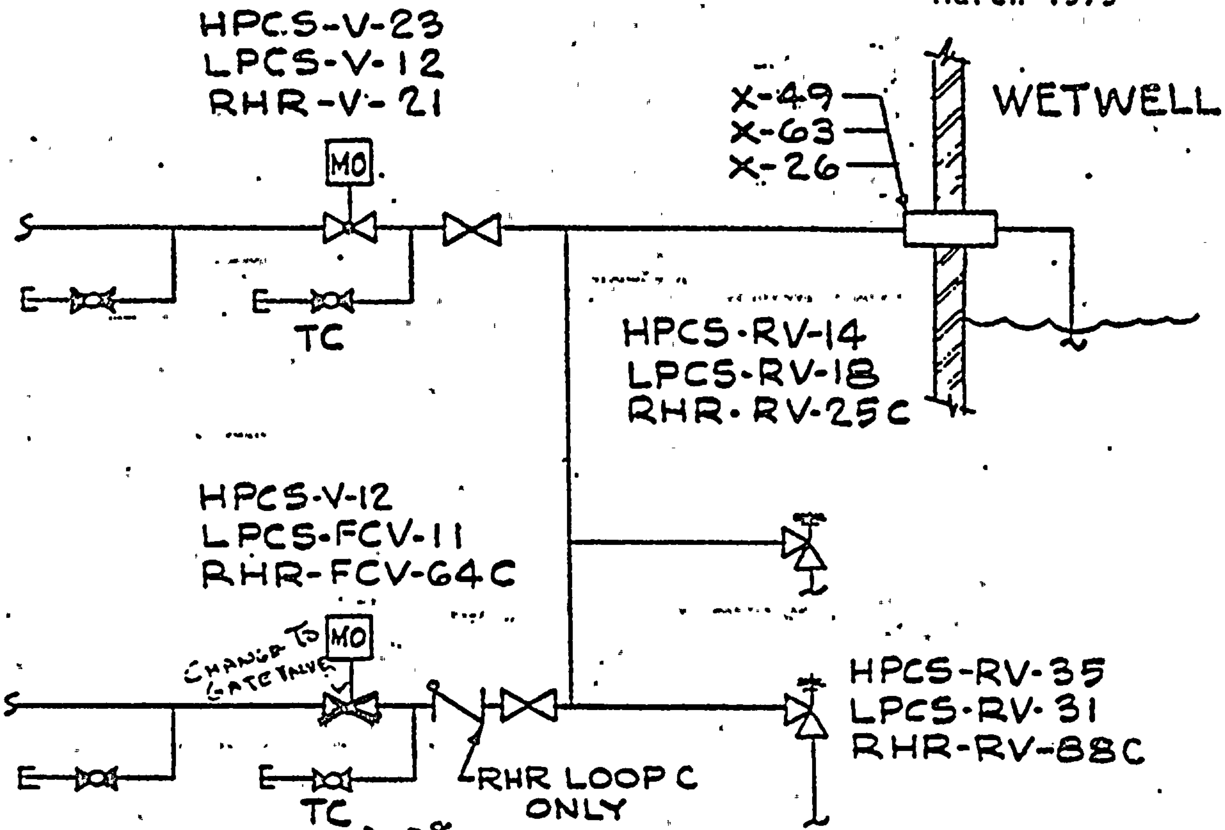


NOTES:
RCIC-V-66 WILL BE "BENCH TESTED" ONCE THE LINE IS REMOVED FOR REFUELING. RHR-V-23 AND RCIC-V-13 CAN BE TESTED ONCE THE FLANGED CONNECTION IS BLANKED OFF AS PER NOTE 1 ON FIG. 6.2-31a. Test Connections Will Be Type C Tested Per Note 8 on FIG. 6.2-31a

RCIC/RHR HEAD SPRAY

Amendment No. 3
March 1979

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

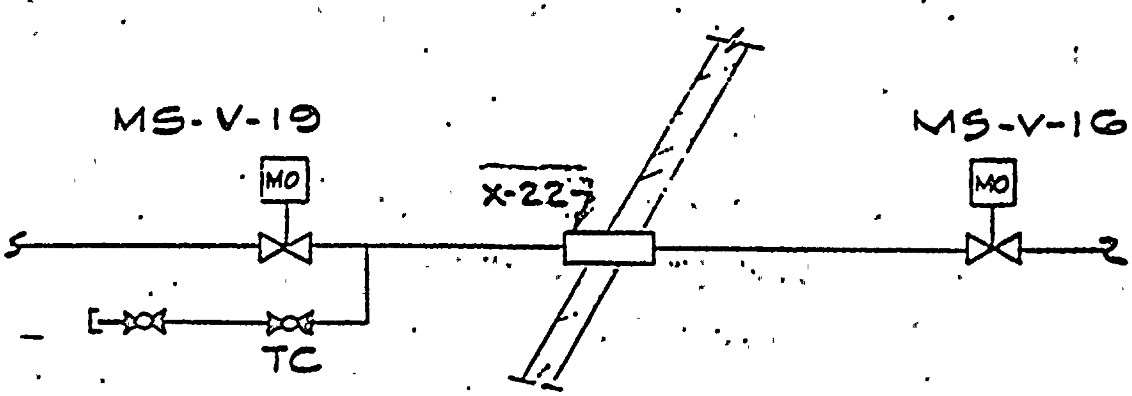


NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

X-49 HPCS TEST LINE

X-63 LPCS TEST LINE

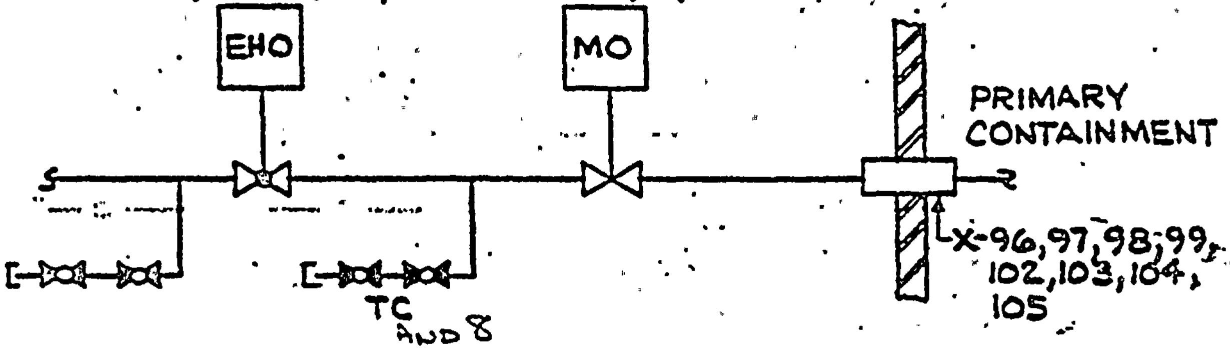
X-26 RHR LOOP C TEST LINE



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

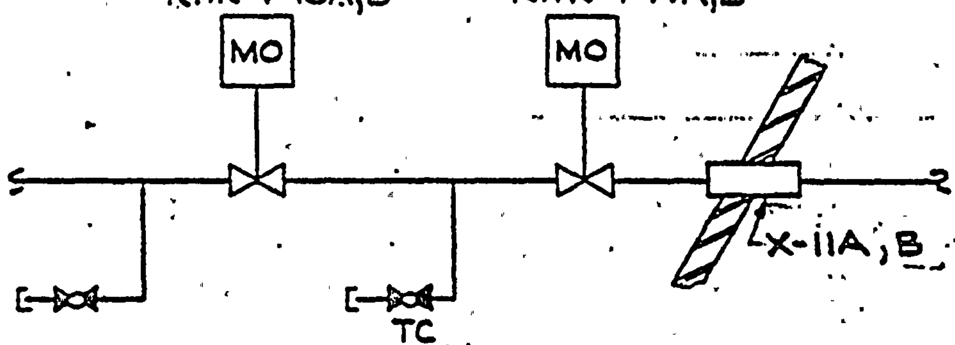
MS DRAIN LINE

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



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

CAC SYSTEM

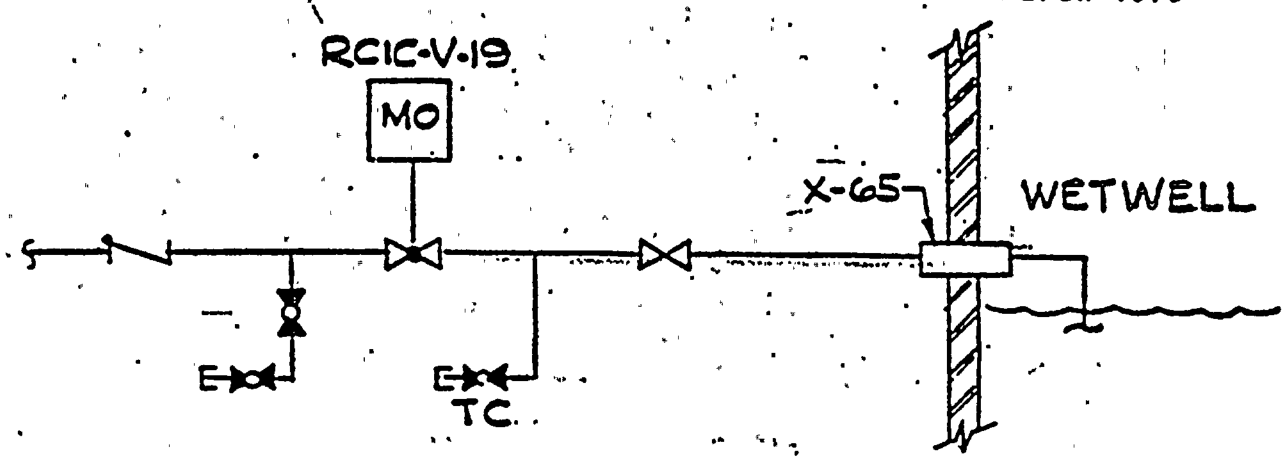


NOTE: SEE NOTES 4 AND 8 ON FIG 6.2-31a

RHR DRYWELL SPRAY

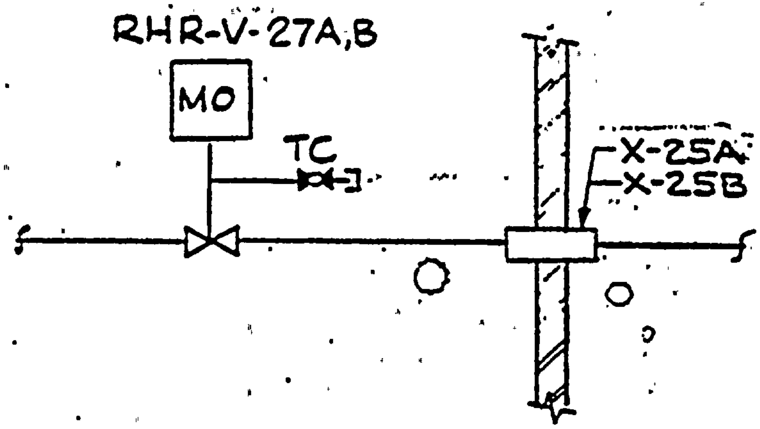
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-96, X-97, X-98, X-99, X-102, X-103, X-104, X-105, X-11A & X-11B	FIGURE 6.2-31g

Amendment No. 3
March 1979



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

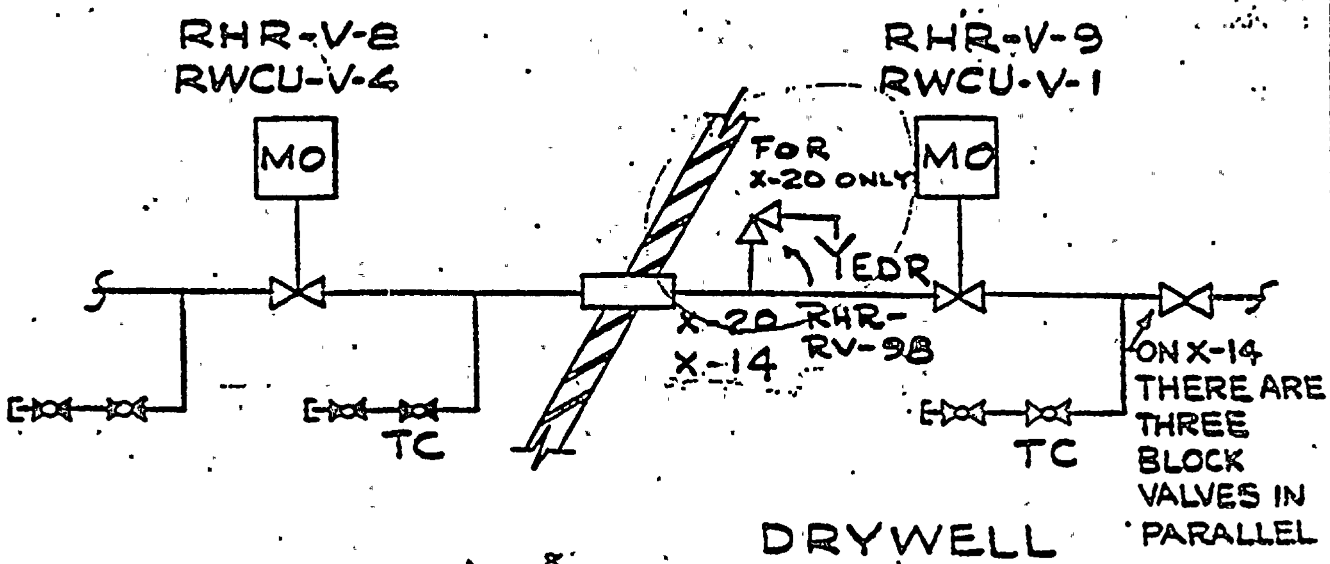
RCIC PUMP MIN. FLOW



NOTE: SEE NOTES 2 AND 8 ON FIG. 6.2-31a

RHR WETWELL SPRAY

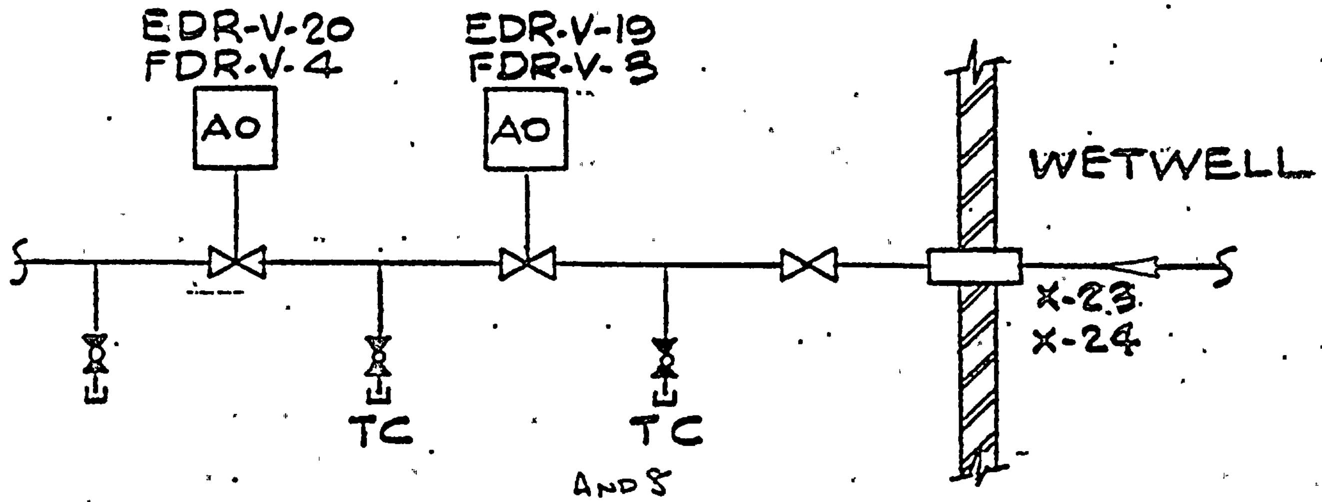
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-65, X-25A & X-25B	FIGURE 6.2-31h



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

X-20 RHR SHUTDOWN COOLING SUPPLY

X-14 RWCU SUCTION



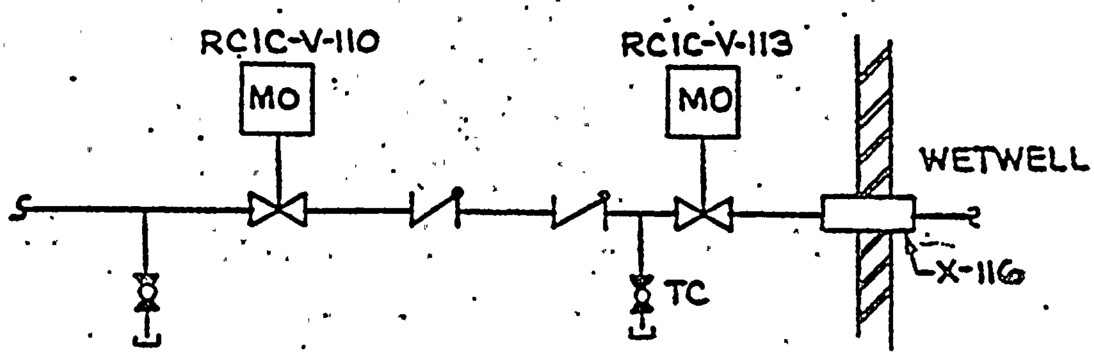
NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a.

X-23 EDR FROM PRIMARY CONTAINMENT

X-24 FDR FROM PRIMARY CONTAINMENT

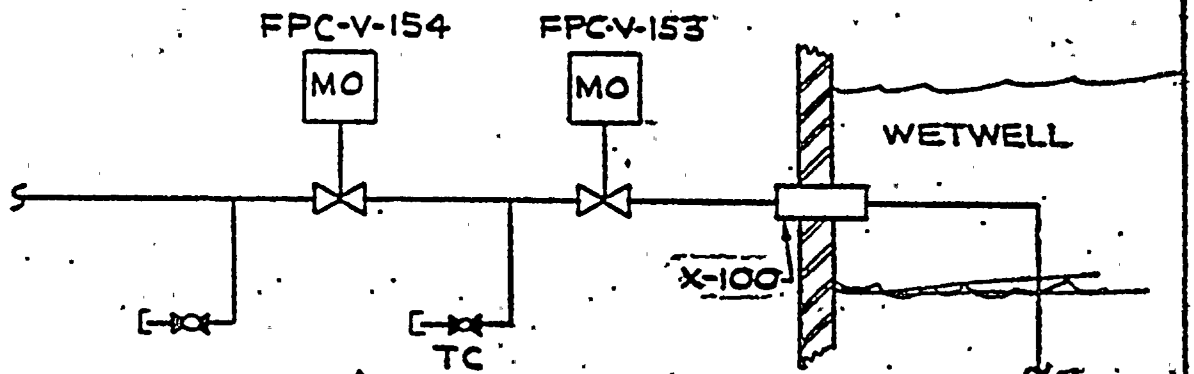
Amendment No. 9
April 1980

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



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

RCIC TURBINE EXHAUST VACUUM BREAKER

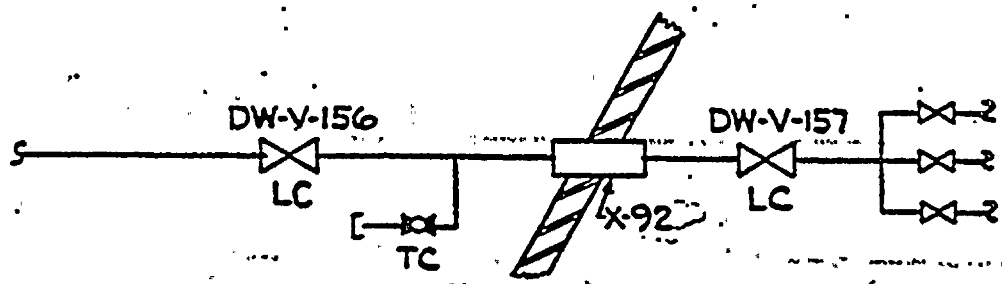


NOTE: SEE NOTES 4 AND 8 ON FIG 6.2-31a

SUPPRESSION POOL CLEAN-UP SUCTION LINE

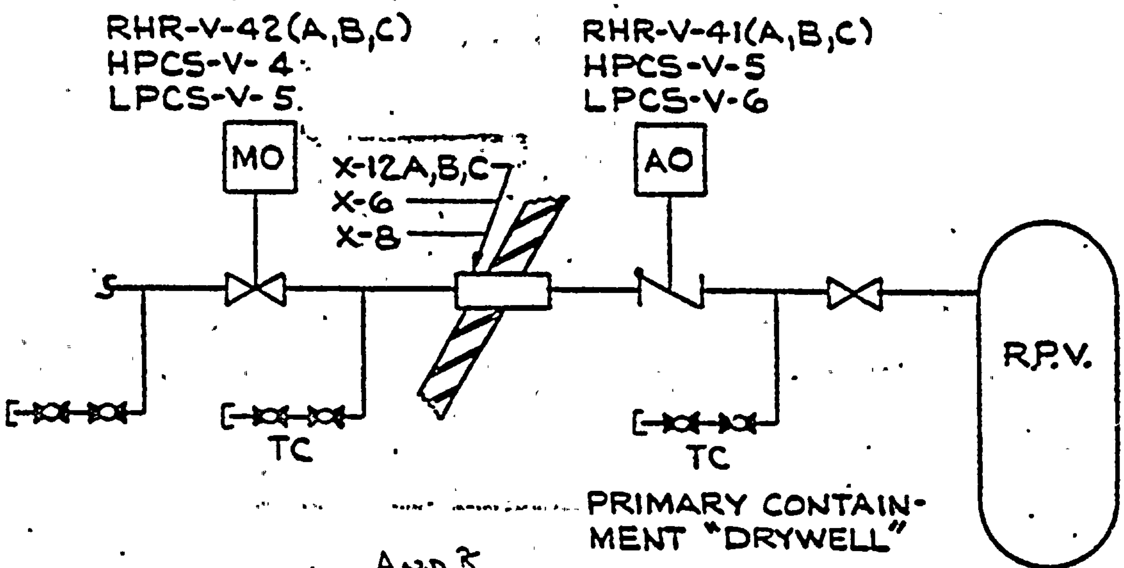
SHOW WATER LEVEL ABOVE PENETRATION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-116 & X-100	FIGURE 6.2-31 i



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

DW SYSTEM



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

X-12A RHR LOOP A LPCI TO RPV

X-12B RHR LOOP B LPCI TO RPV

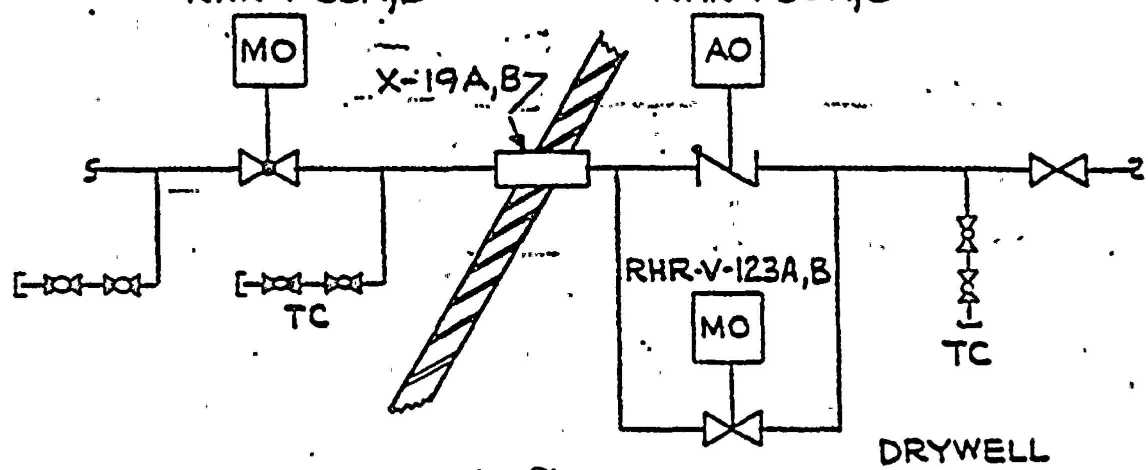
X-12C RHR LOOP C LPCI TO RPV

X-6 HPCS TO RPV

X-8 LPCS TO RPV

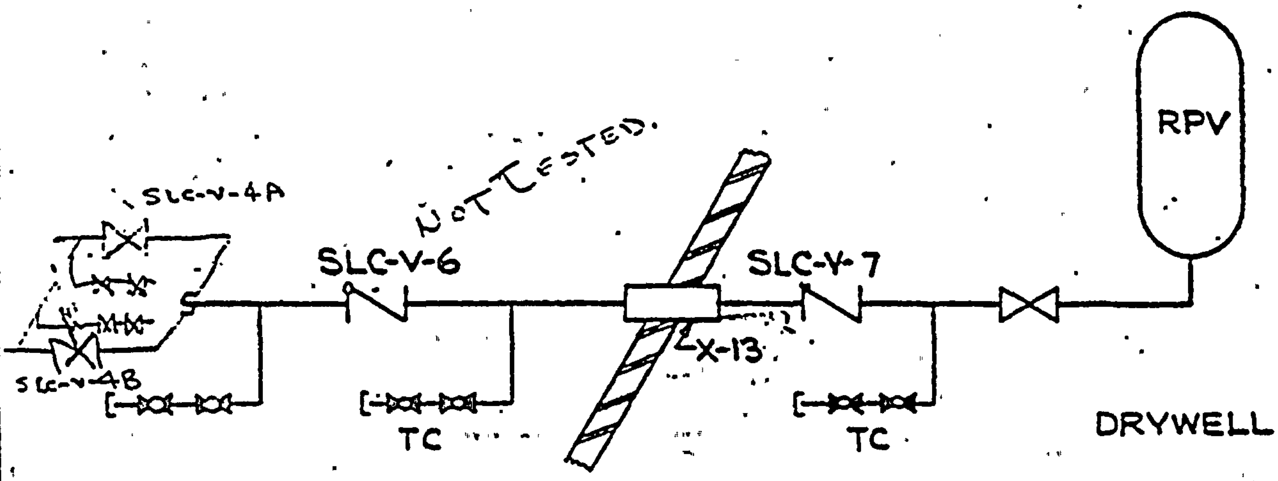
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-92, X-12A, X-12B, X-12C, X-6 & X-8	FIGURE 6.2-31e

Amendment No. 3
March 1979



NOTE: SEE NOTES 1 AND 8 ON FIG. 6.2-31a

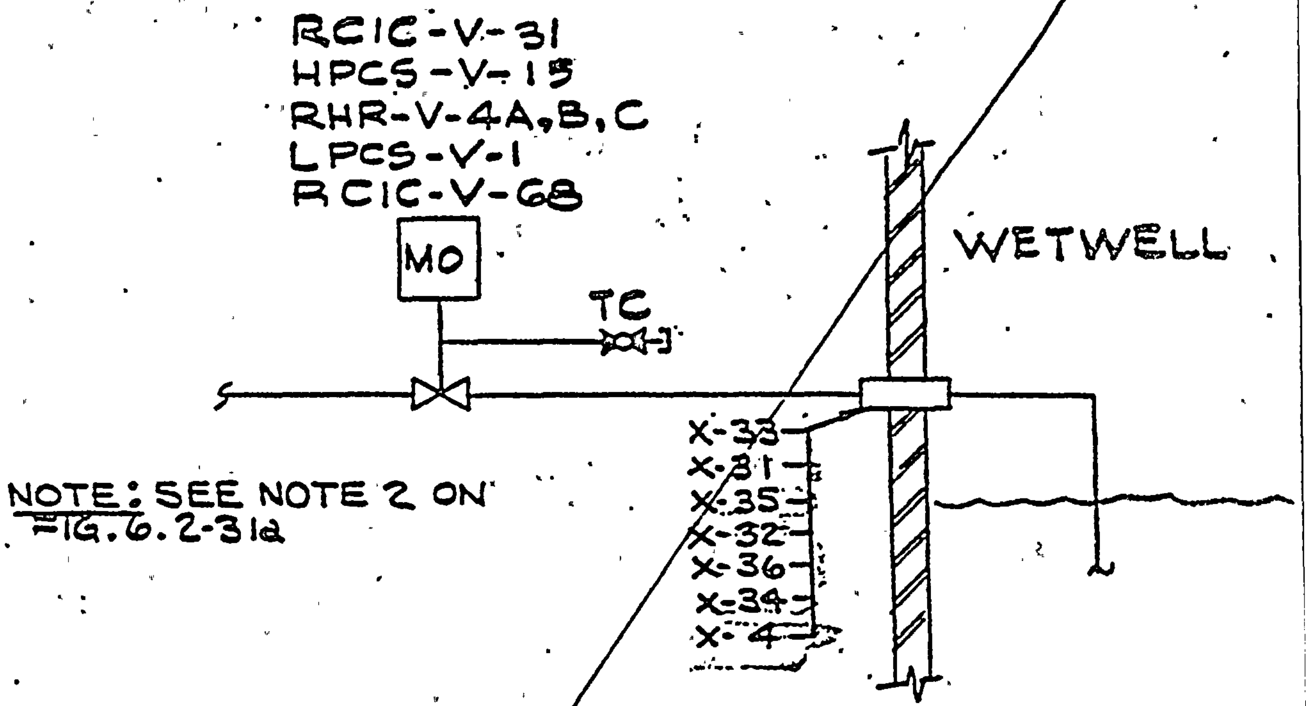
RHR SHUTDOWN COOLING RETURN



NOTE: SEE NOTES 1 AND 8 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



X-33 RCIC PUMP SUCTION FROM SUPPRESSION POOL

X-31 HPCS PUMP SUCTION FROM SUPPRESSION POOL

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

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

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

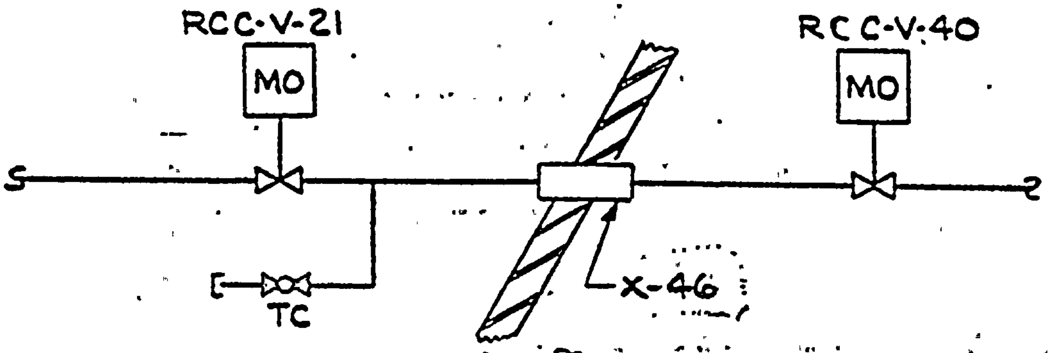
X-34 LPCS PUMP SUCTION FROM SUPPRESSION POOL

X-4 RCIC TURBINE EXHAUST

To Be Revised

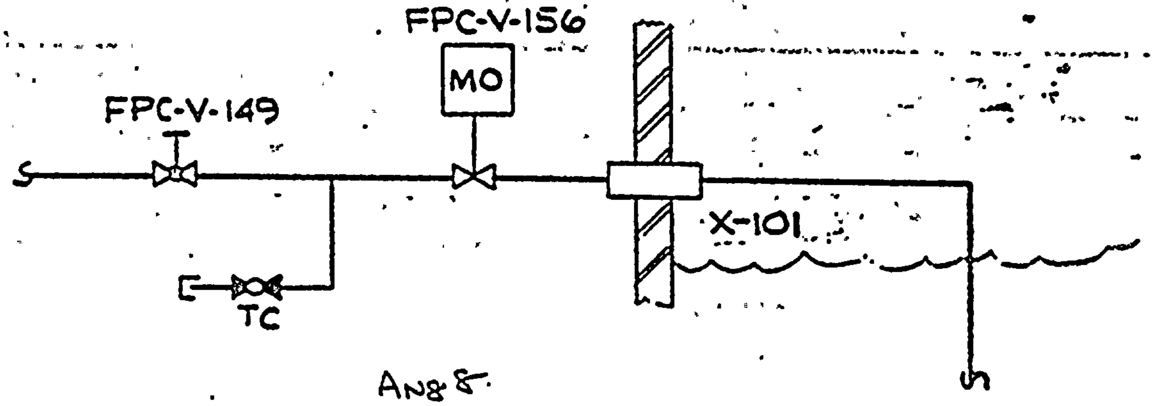
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-33, X-31, X-35, X-32, X-36, X-34, X-4	FIGU[RE] 6.2-3

Amendment No. 3
March 1979



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

RCC RETURN LINE

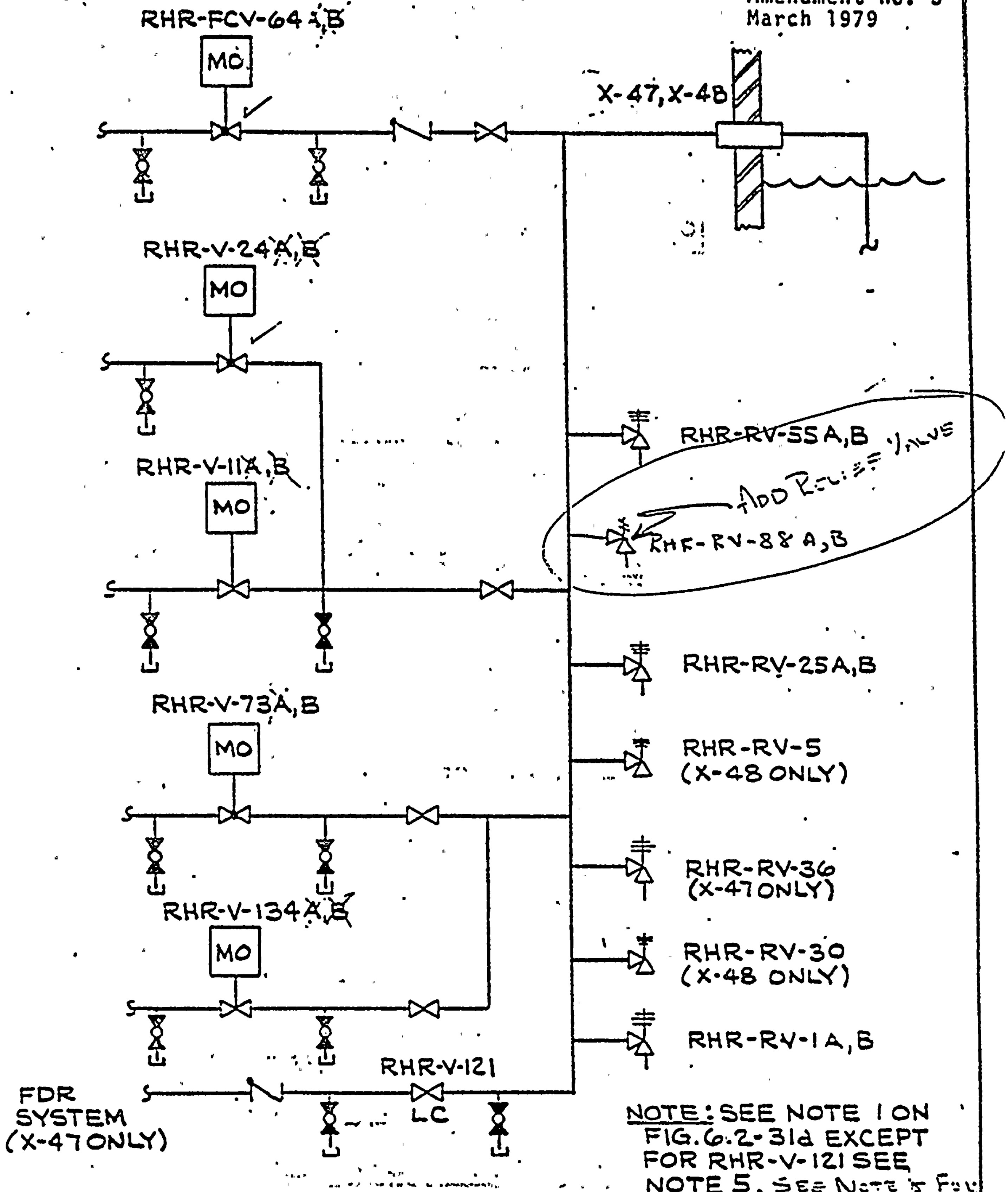


NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

SUPPRESSION POOL CLEAN-UP RETURN LINE

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

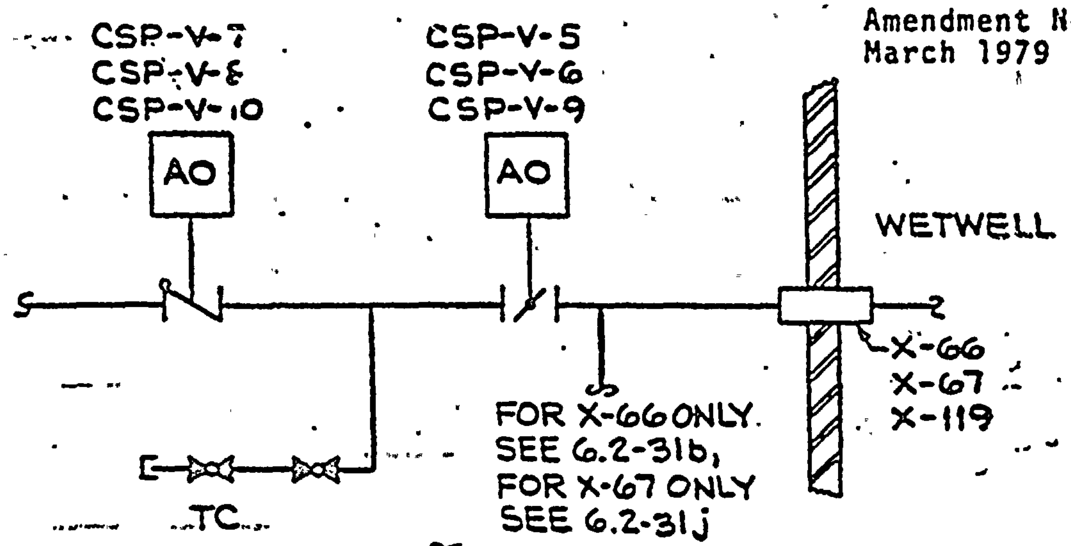
Amendment No. 3
March 1979



RHR COMBINED RETURN LINE
TO SUPPRESSION POOL

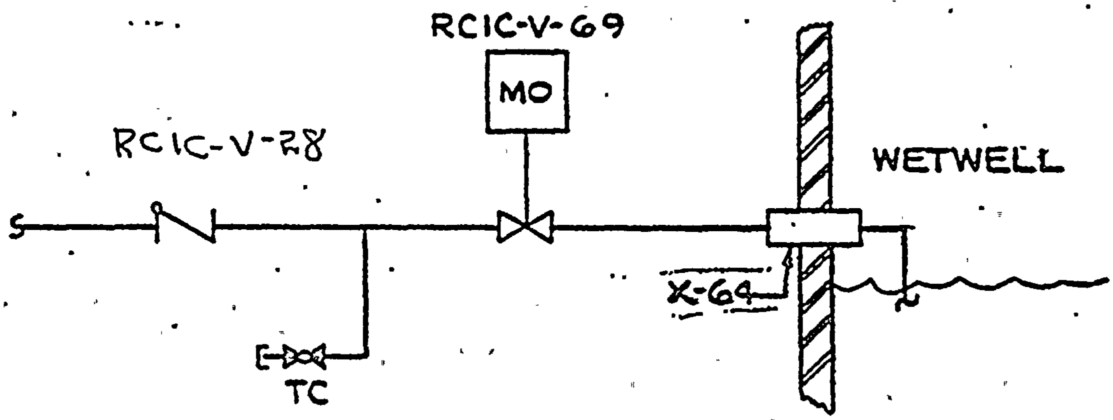
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-47 AND X-48	FIGURE 6.2-31p

Amendment No. 3
March 1979



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

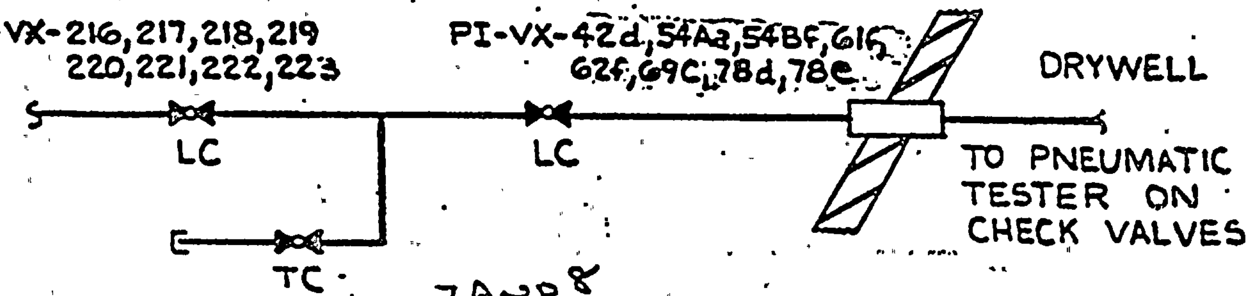
REACTOR BUILDING TO WETWELL VACUUM RELIEF



NOTE: SEE NOTE ~~8~~ 4 ON FIG. 6.2-31a

RCIC VACUUM PUMP DISCHARGE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-66, X-67, X-119 & X-64	FIGURE 6.2-31q



NOTE: SEE NOTES ~~4~~ 7 AND 8 ON FIGURE 6.2-31a.

Note 4 Does Not Apply

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 ✓

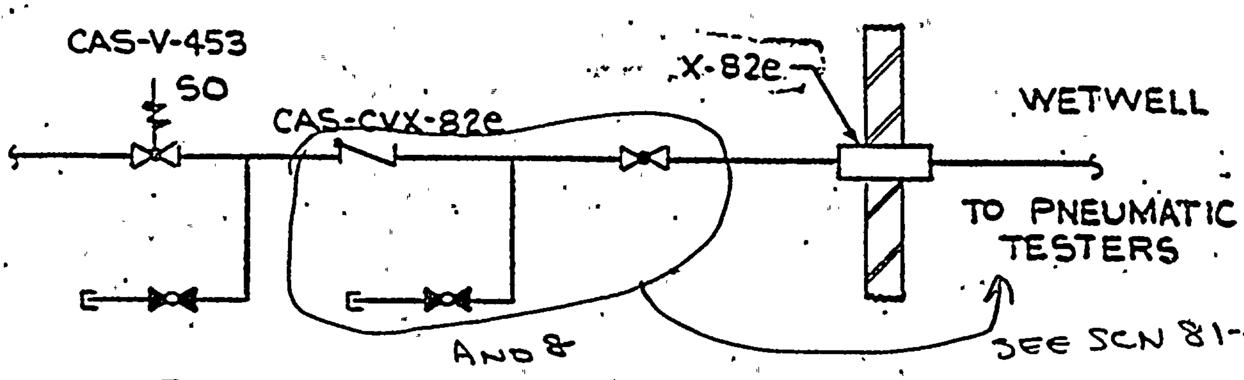
X-61f AIR LINE FOR TESTING RHR-V-41A ✓

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 ✓

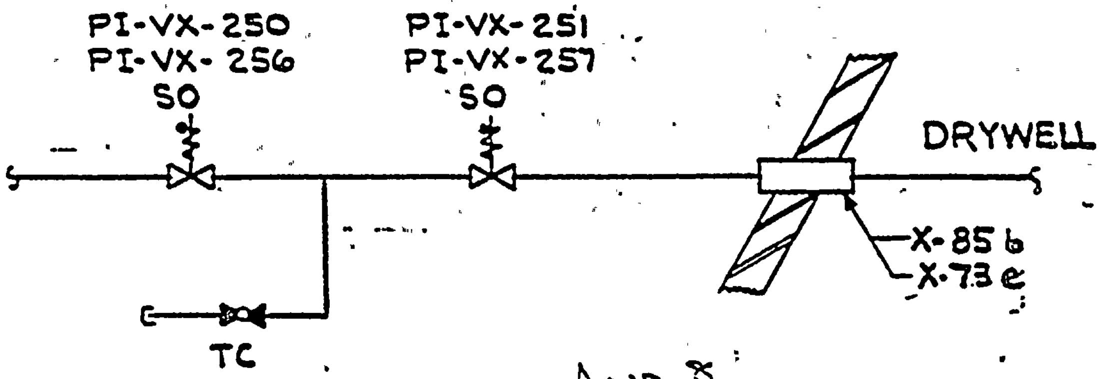


NOTE: SEE NOTES 1 AND 8 ON FIGURE 6.2-31a

AIR LINE FOR TESTING WETWELL TO DRYWELL VACUUM BREAKERS

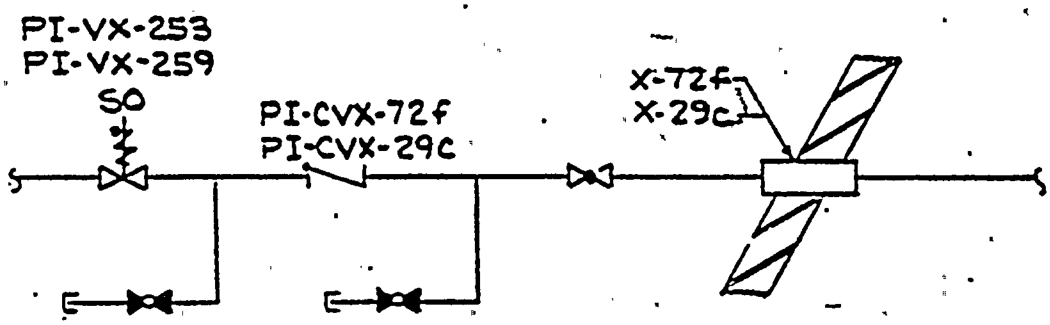
Amendment No. 3
March 1979

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 NOTES 4 AND 8 ON FIGURE 6.2-31a

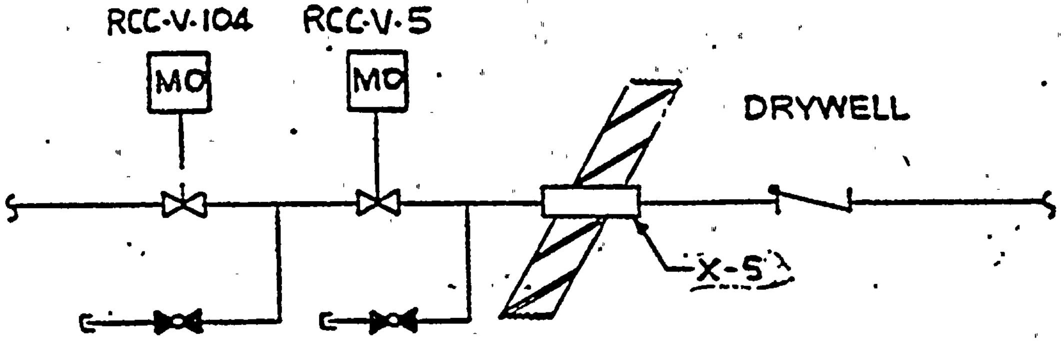
RADIATION MONITOR SUPPLY LINE DIVISION A
RADIATION MONITOR SUPPLY LINE DIVISION B



NOTE: SEE NOTES 1 AND 8 ON FIGURE 6.2-31a

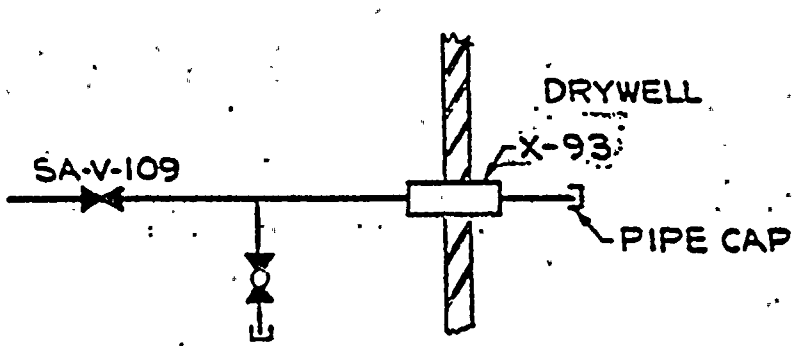
RADIATION MONITOR RETURN LINE DIVISION A
RADIATION MONITOR RETURN LINE DIVISION B

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-85b, X-73e, X-72f, & X-29c	FIGUR 6.2-3



NOTE: SEE NOTES 4 AND 8 ON FIGURE 6.2-31a

RCC SUPPLY LINE



SERVICE AIR FOR MAINTENANCE

NOTE: SEE NOTE 8 ON FIGURE 6.2-31a

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATION X-5 & X-93	FIGURE 6.2-31c

CONTAINMENT SYSTEMS BRANCH

ISSUE 9

NRC:	Is it correct to show LPCS minimum flow lines open during normal operation?
Supply System:	The Supply System will revise Table 6.2-16 to indicate these valves are closed during normal operation.
	See revised FSAR page 6.2-122 (attached).

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
HPCS to Reactor	6	3.2-7 6.2-31L	55	A	HPCS-V-5	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					HPCS-V-4	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	17	9						
LPCS to Reactor	8	3.2-7 6.2-31L	55	A	LPCS-V-6	Check	I	Process	Process	-	-	C	C	O/C	-	12	-	-	Yes	W	Valves	R.B.	No	3, 24
					LPCS-V-5	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	12	27	22						
HPCS pump suction from suppression pool	31	3.2-7 6.2-31a	56	B	HPCS-V-15	MO Gate	O	AC	AC	46	Manual	C	C	O/C	AS-IS	18	18	3	Yes	W	Valves	R.B.	No	18 24
LPCS pump suction	34	3.2-7 6.2-31a	56	B	LPCS-V-1	MO Gate	O	AC	AC	46	Manual	O	O	O/C	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18 24
HPCS test line	49	3.2-7 6.2-31f	56	B	HPCS-V-23	MO Globe	O	AC	AC	F,A	RM	C	C	C	AS-IS	12	Std	6	Yes	W	Valves	R.B.	No	18
HPCS pump min. flow					HPCS-V-12	MO Gate	O	AC	AC	38	RM	C	C	O/C	AS-IS	4	4	53						
HPCS suction relief					HPCS-RV-14	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	65						19
HPCS discharge relief					HPCS-RV-35	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	70						19
LPCS test line	63	3.2-7 6.2-31f	56	B	LPCS-V-12	MO Globe	O	AC	AC	F,Y	RM	C	C	C	AS-IS	12	Std	4	Yes	W	Valves	R.B.	No	18
LPCS pump min. flow					LPCS-~~V~~FCV-11	MO Globe	O	AC	AC	38	RM	X C	C	O/C	AS-IS	3	Std	87						
LPCS suction relief					LPCS-RV-31	Relief	O	PP	Spring	-	-	C	C	C	-	1	-	25						19
LPCS discharge relief					LPCS-RV-18	Relief	O	PP	Spring	-	-	C	C	C	-	2	-	50						19
SLC to Reactor	13	3.2-5 6.2-31m	55	A	SLC-V-7	Check	I	Process	Process	-	-	C	C	O	-	1-1/2	-	-	No	W	Valves	R.B.	No	
					SLC-V-6	Check	O	Process	Process	-	-	C	C	C	-	1-1/2	-	6						
					SLC-V-4A	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21
					SLC-V-4B	Explosive	O	AC		-	-	C	C	C	-	1-1/2	-	136						21

CONTAINMENT SYSTEMS BRANCH

ISSUE 11

NRC:	Reactor recirculation hydraulic line penetrations X-76, X-77 are to meet GDC-56 to GDC-57 as indicated in the FSAR. Justify why WNP-2 does not supply two isolation valves. It is not an ESF system and cannot be considered a closed system inside or outside containment.
Supply System:	Second isolation valve added outboard per letter, WPBR-RO-81-183, dated October 12, 1981, and the attached FSAR page changes.

6.2.4.3.2.2.3.5 Reactor Building to Wetwell (RB-WW) Vacuum Relief Lines

The RB-WW vacuum relief penetrations, three in total, are each equipped with a positive closing swing check valve in series with an air-operated butterfly valve. The air operator on the swing check valve is used only for testing. The air operated butterfly valve is controlled by a differential pressure indicating switch which senses the pressure difference between the suppression chamber and the reactor building. When the negative pressure in the suppression chamber exceeds the instrument setpoint, the butterfly valve opens. The arrangement of valves and instruments is shown in Figure 3.2-15. See Table 6.2-15 for differential pressure indicating switch characteristics.

6.2.4.3.2.2.3.6 Reactor Recirculation (RRC) Flow Control Valve Hydraulic Lines

6.2.4.3.2.2.4 Conclusion on Criterion 56

In order to assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure retaining components of these systems are designed to the same quality standards as the containment.

6.2.4.3.2.3 Evaluation Against Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 govern comprise the closed system isolation valve group.

Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as closely as possible to the containment boundary.

TIP subsystem guide tubes are provided with an isolation valve which closes automatically upon receipt of a proper signal and after the TIP cable and fission chamber have been retracted. In series with this isolation valve is included an additional or backup isolation shear valve. Both valves are located outside the drywell. The TIP system and isolation provisions are discussed in Note 29 of Table 6.2-16.

The ~~RRC system~~ four hydraulic control lines to each of the RRC system ~~the~~ flow control valves contain ~~an~~ Two isolation valves per line located outside the drywell. Both isolation valves are solenoid-operated and ~~which~~ closes automatically upon receipt of ~~an~~ an isolation signal. The hydraulic lines and their isolation valves are discussed in Note 28 of Table 6.2-16.

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
RFW Line A	17A	3.2-2 6.2-31b	55	A	RFW-V-10A	Check	I	Process	Process	-	-	O	O/C	O/C	-	24	-	-	No	W	Valves	T.B.	1.5	16
					RFW-V-32A	PC Check	O	Process	Pro/Spr	-	-	O	O/C	O/C	-	24	-	2	No	W				
					RFW-V-65A	MO Gate	O	AC	AC	31	Manual	O	O/C	O/C	AS-IS	24	Std	8	No	W				
					RWCU-V-40	MO Gate	O	AC	AC	47	Manual	O	O	C	AS-IS	6	Std	24	No	W				
RFW Line B	17B	3.2-2 6.2-31b	55	A	RFW-V-10B	Check	I	Process	Process	-	-	O	O/C	O/C	-	24	-	-	No	W	Valves	T.B.	1.5	16
					RFW-V-32B	PC Check	O	Process	Pro/ Spr	-	-	O	O/C	O/C		24		2	No	W				
					RFW-V-65B	MO Gate	O	AC	AC	31	Manual	O	O/C	O/C	AS-IS	24	Std	8	No	W				
					RWCU-V-40	MO Gate	O	AC	AC	47	Manual	O	O	C	AS-IS	6	Std	24	No	W				
RRC Hydraulic Lines		3.2-3	~~57~~ 56	B																				
Cylinder	76f				HY-V-33A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	5	No	H	Valves	R.B.	No	28
					HY-V-33B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	6						
Cylinder	76b				HY-V-34A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	5						
					HY-V-34B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	6						
Shuttle	76e				HY-V-35A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	5						
					HY-V-35B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	6						
Drain	76c				HY-V-36A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	5						
					HY-V-36B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	6						
Cylinder	77f				HY-V-17A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	5	No	H	Valves	R.B.	No	28
					HY-V-17B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	6						
Cylinder	77b				HY-V-18A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	5						
					HY-V-18B	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	3/4	<5	6						
Shuttle	77e				HY-V-19A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	5						
					HY-V-19A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	6						
Drain	77c				HY-V-20A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	5						
					HY-V-20A	SO Globe	O	AC	Spring	A,F	RH	O	O	C	C	1/2	<5	6						

TABLE 6.2-16 (Continued)

indication lights which can alert the operators to the fact that the check valve is not fully closed. The operator can then remotely shut the valve by means of a pneumatic operator. The operating switch is spring-return to neutral so the vacuum breaker function will not be impaired. The air supply to these valves is Quality Class I.

27. Instrument lines that penetrate primary containment conform to Regulatory Guide 1.11. The lines that connect to the reactor pressure boundary include a restricting orifice inside containment, are Seismic Category I and terminate in instruments that are Seismic Category I. The instrument lines also include manual isolation valves and excess flow check valves or equivalent (see hydrogen monitor return lines). These penetrations will not be type C tested since the integrity of the lines are continuously demonstrated during plant operations where subject to reactor operating pressure. In addition, all lines are subject to the type A test pressure on a regular interval. Leaktight integrity is also verified with completion of functional and calibration surveillance activities as well as by visual inspection during daily operator patrols as applicable.

28. Penetrations X-76 and X-77 contain lines for the hydraulic control of the reactor recirculation flow control valve. These lines contain ~~corrosive~~ hydraulic fluid used to position the reactor recirculation flow control valve.

These lines inside of the containment are Seismic Category I and Quality Group B. Each line is ~~They are~~ provided with two fail~~ed~~-closed solenoid-operated ~~automatic~~ isolation valves ~~outside the containment~~ which receive an automatic isolation signal on high drywell pressure or reactor vessel low water level.

~~These lines meet the requirement of General Design Criterion 57 and therefore require only single automatic isolation valves outside of the containment. These lines also meet the requirement of Standard Review Plan 6.2.4. They are designed to Seismic Category I, Code Group B and the following criteria:~~

Both isolation valves are located outside containment to improve valve reliability because of more favorable environmental conditions (i.e., potential damage to the solenoid valves resulting from humidity, radiation, pressure and temperature transients, and post-LOCA pipe whip and jet impingement is greatly reduced). Also, this location allows for ease of maintenance and manual override operation, if required.

TABLE 6.2-16 (Continued)

~~a. do not communicate with either the reactor coolant system or the containment atmosphere,~~

~~b. are protected against missiles and pipe whip,~~

~~c. are designed to withstand temperatures at least equal to the containment design temperature,~~

~~d. are designed to withstand the external pressure from the containment structural acceptance test, and~~

~~e. are designed to withstand the loss-of-coolant accident transient and environment.~~

~~Even if the failed closed valve were to not shut there will be no leakage of containment atmosphere through the hydraulic control lines since the piping inside the primary containment remains intact. There are no active component failures which would compromise the integrity of the closed system inside the primary containment.~~ Integrity of the ~~closed~~ system inside the primary containment is, essentially, continuously ~~constantly~~ monitored since the system is under a constant operating pressure of 1800 psig. Any leakage through this system would be noticed because of ~~operation would be~~ erratic operation and because of alarms for abnormal operation ~~indications~~ provided on the hydraulic control unit. ~~In addition,~~

New P In order to perform type C tests on these lines, the system would have to be disabled and drained of the corrosive hydraulic fluid. This is considered to be detrimental to the proper operation of the system in that possible damage could occur in establishing the test condition or restoring the system to normal.

For these reasons, the ~~These~~ lines and associated isolation valves Are ~~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.

SYSTEM	INBOARD ISOLATION VALVE	OUTBOARD ISOLATION VALVE	CLASS.	ISOLATION SIGNALS LOCA	ISOLATION SIGNALS SYSTEM	COMMENTS
MAIN STEAM						
- main steam lines	MS-V-22A,B,C,D	MS-V-28A,B,C,D MS-V-67A,B,C,D	NE	A	C,G,D,P	
- MSIV-leakage control	MS-V-22A,B,C,D	MSLC-V-3A,B,C,D	E	-		Leakage control for MSIV.
- MS line drain	MS-V-16	MS-V-19	NE	A	C,G,D,P	
RRC						
- hydraulic lines	HY-V-17A,18A,19A, 20A,33A,34A,35A,36A	HY-V-17B,18B,19B, 20B,33B,34B,35B,36B	NE	A,F		
- pump seal water	RRC-V-13A,B	RRC-V-16A,B	E			Valves must stay open to prevent reactor coolant loss through seals.
HPCS						
- to RPV	HPCS-V-5	HPCS-V-4	E			Essential safety system.
- suppression pool suction	-	HPCS-V-15	E			
- test line	-	HPCS-V-23	NE	F,A		
- minimum flow line	-	HPCS-V-12	E			
LPCS						
- to RPV	LPCS-V-6	LPCS-V-5	E			Essential safety system.
- suppression pool suction	-	LPCS-V-1	E			
- test line	-	LPCS-V-12	NE	F,V		
- minimum flow line	-	LPCS-FCV-11	E			

CONTAINMENT SYSTEMS BRANCH

ISSUE 12

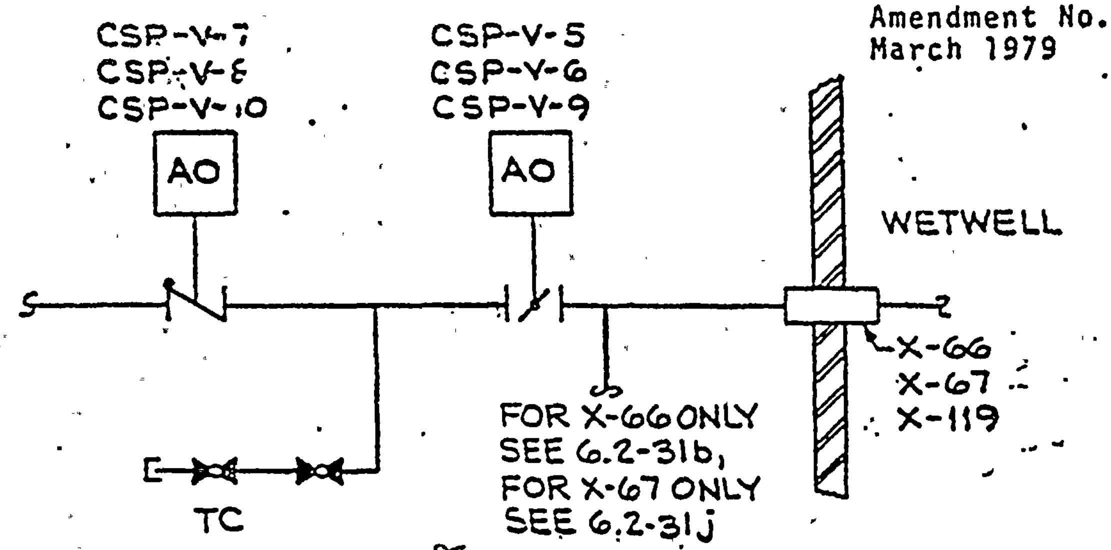
NRC: Page 6.2-123, RCIC turbine exhaust has only one valve shown. Add the check valve (RCIC-V-40) for containment isolation purposes.

Supply System: The Supply System will revise page 6.2-123 to indicate that the check valve is a containment isolation valve.

Check valve RCIC-V-28 in the Vacuum Pump Discharge Line (Penet. 64) must also be included as an isolation valve.

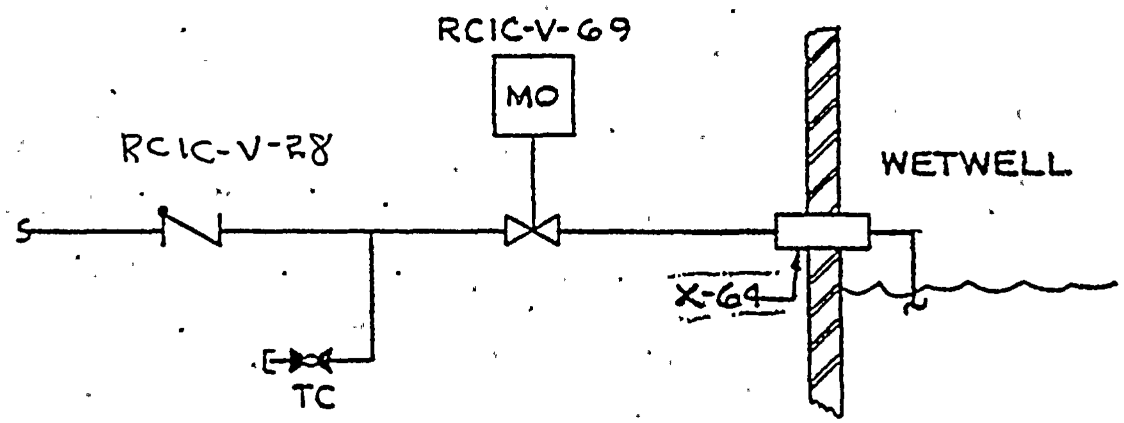
TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Note
DW Service Line	92	9.2-4 6.2-31L	56	B	DW-V-157	Gate	I	Manual	Manual	-	-	LC	LC	LC	-	2	-		No	W	Valves	S.B.	.13	
					DW-V-156	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	5						
RHR Condensing Mode Steam Supply	21	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	Yes	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-64	MO Gate	O	DC	DC	X	RM	C	C	C	AS-IS	10	16	2						
RCIC Turbine Steam Supply	45	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	No	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-8	MO Gate	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4	Std	2						
RCIC Pump Minimum Flow	65	3.2-8 6.2-31h	56	B	RCIC-V-19	MO Globe	O	DC	DC	33	RM	C	C	O/C	AS-IS	2	5	7	No	W	Valves	R.B.	No	22
RCIC Turbine Exhaust	4	3.2-8 6.2-31n	56	B	RCIC-V-68	MO Gate	O	DC	DC	35	Manual	O	O	O/C	AS-IS	10	Std	10	No	S	Valves	R.B.	No	22
			56	B	RCIC-V-40	CHECK	O	Process	Process	-	-	O	C	O/C	-	10	-	12	NO	S	VALVES	R.B.	NO	
RCIC Turbine Exhaust Vacuum Breaker	116	3.2-8 6.2-31l	56	B	RCIC-V-110	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	9	No	A	Valves	R.B.	No	17
					RCIC-V-113	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	5						
RCIC Vacuum Pump Discharge	64	3.2-8 6.2-31q	56	B	RCIC-V-69	MO Gate	O	DC	DC	36	Manual	O	O	O/C	AS-IS	1-1/2	Std	4	No	W	Valves	R.B.	No	22
			56	B	RCIC-V-28	CHECK	O	Process	Process	-	-	O	C	O/C	-	1½	-	6	NO	W	VALVES	R.B.	No	
RCIC Pump Suction from Suppression Pool	33	3.2-8 6.2-31n	56	B	RCIC-V-31	MO Gate	O	DC	DC	32	Manual	C	C	O/C	AS-IS	8	Std	2	No	W	Valves	R.B.	No	23
RPV Head Spray	2	3.2-8 6.2-31e	55	A	RCIC-V-66	Check	I	Process	Process	-	-	C	O	O/C	-	6	-	-	No	W	Valves	R.B.	No	3
					RCIC-V-13	MO Gate	O	DC	DC	34	RM	C	O/C	O/C	AS-IS	6	15	2	No	W	Valves	R.B.	No	
					RHR-V-23	MO Globe	O	DC	DC	L,U,H,R	RM	C	O/C	C	AS-IS	6	Std	7	Yes	W	Valves	R.B.	No	



NOTE: SEE NOTES 4 AND 8 ON FIG. 6.2-31a

REACTOR BUILDING TO WETWELL VACUUM RELIEF



NOTE: SEE NOTE ~~5~~ 4 ON FIG. 6.2-31a

RCIC VACUUM PUMP DISCHARGE

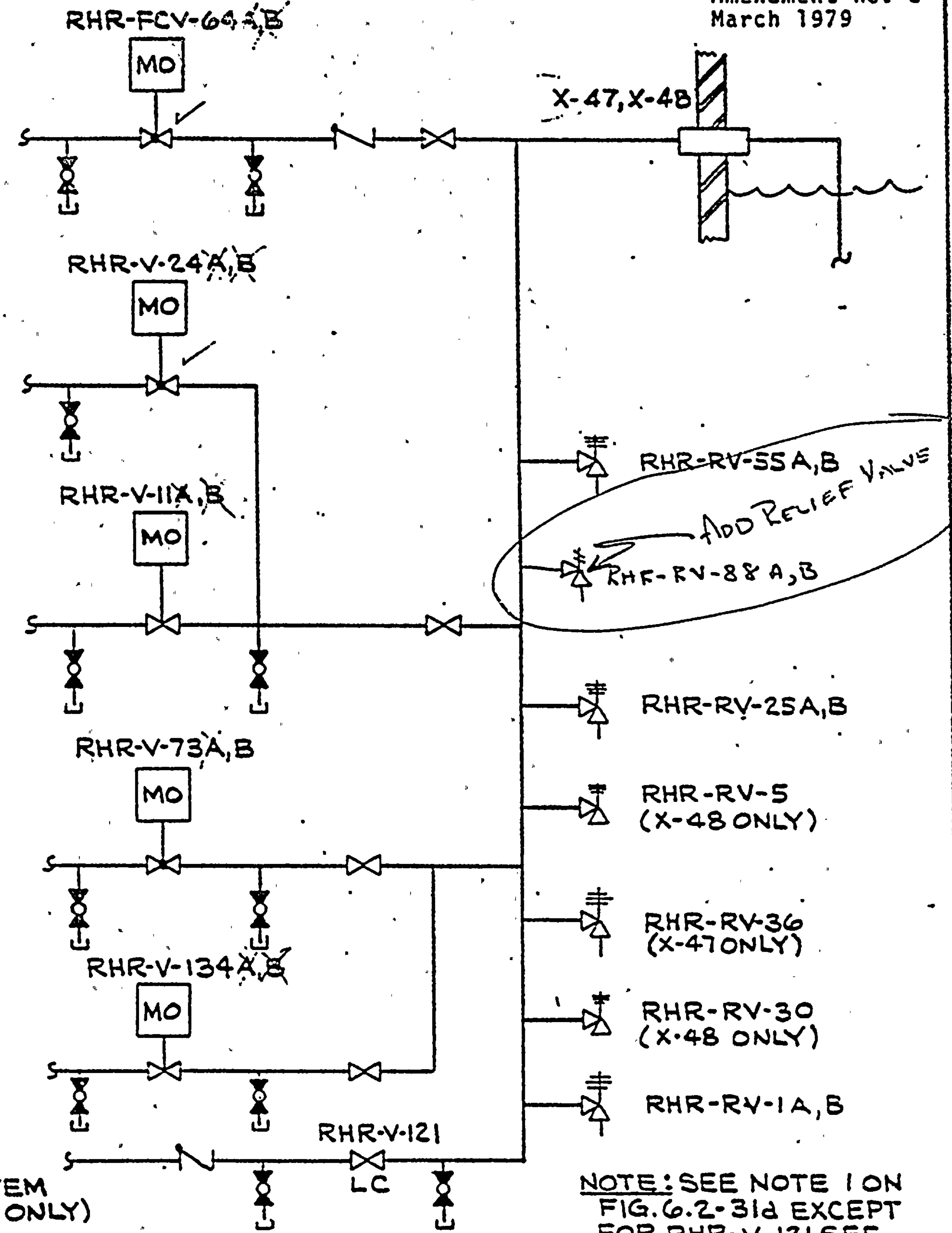
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS X-66, X-67, X-119 & X-64	FIGURE 6.2-31q

CONTAINMENT SYSTEMS BRANCH

ISSUE 15

NRC:	Page 6.2-125, RHR-88a, pump suction relief valve, is not shown in Figure 6.2-31p.
Supply System:	Figure 6.2-31p will be revised accordingly
	See revised FSAR Figure 6.2-31p (attached).

Amendment No. 3
March 1979



NOTE: SEE NOTE 1 ON FIG. 6.2-31d EXCEPT FOR RHR-V-121 SEE NOTE 5. See Note 5 For Testing Of Test Connections

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

CONTAINMENT SYSTEMS BRANCH

ISSUE 16

NRC:	Add Seismic Quality Class 1 debris screens to vacuum breaker.
Supply System:	The Supply System will install by Fuel Load per issue number 1 statement.

See revised FSAR page 6.2-66 (attached).

6.2.4.3.2.2.3.5 Reactor Building to Wetwell (RB-WW) Vacuum Relief Lines

The RB-WW vacuum relief penetrations, three in total, are each equipped with a positive closing swing check valve in series with an air-operated butterfly valve. The air operator on the swing check valve is used only for testing. The air operated butterfly valve is controlled by a differential pressure indicating switch which senses the pressure difference between the suppression chamber and the reactor building. When the negative pressure in the suppression chamber exceeds the instrument setpoint, the butterfly valve opens. The arrangement of valves and instruments is shown in Figure 3.2-15. See Table 6.2-15 for differential pressure indicating switch characteristics.

SEISMIC CATEGORY I STAINLESS STEEL GRILLES ARE INSTALLED ACROSS THE SWING CHECK VALVE OPENINGS TO PROHIBIT DEBRIS FROM ENTERING AND PREVENTING THE VALVES FROM SEATING.

6.2.4.3.2.2.4 Conclusion on Criterion 56

In order to assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure retaining components of these systems are designed to the same quality standards as the containment.

6.2.4.3.2.3 Evaluation Against Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 govern comprise the closed system isolation valve group.

Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as closely as possible to the containment boundary.

TIP subsystem guide tubes are provided with an isolation valve which closes automatically upon receipt of a proper signal and after the TIP cable and fission chamber have been retracted. In series with this isolation valve is included an additional or backup isolation shear valve. Both valves are located outside the drywell. The TIP system and isolation provisions are discussed in Note 29 of Table 6.2-16.

The RRC system hydraulic control lines to the flow control valve contain an isolation valve located outside the drywell which closes automatically upon receipt of its isolation signal. The hydraulic lines and their isolation valves are discussed in Note 28 of Table 6.2-16.

CONTAINMENT SYSTEMS BRANCH

ISSUE 18

NRC: Page 6.2-123, several valves are indicated as manual operating valves. Unless locked closed, this is not acceptable.

Supply System: The Supply System will revise the table (table 6.2-16) to indicate remote, manual operation for valves in question:

RCIC-V-68, 69, 31
RHR-V-16A, B
RHR-V-17A, B
RHR-V42A, B, C
RHR-V-73A
RHR-V-134A
RHR-V-4A, B, C
RHR-V-124A, B
RHR-V-125A, B
CAC-V-2
CAC-FCV-2A
CAC-V-15
CAC-FCV-1B
CAC-V-11
CAC-FCV-2B
CAC-V-6
CAC-FCV-1A
CAC-V-4
CAC-FCV-4A, B
CAC-V-13
CAC-V-17
CAC-FCV-3A, B
CAC-V-8
RRC-V-16A, B
CIA-V-20
CIA-V-20
CIA-V-30A, B

See revised FSAR pages 6.2-123, 124, 125, 126, 127, 129 and 130 (attached).

FSAR Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
DW Service Line	92	9.2-4 6.2-31L	56	B	DW-V-157	Gate	I	Manual	Manual	-	-	LC	LC	LC	-	2	-		No	W	Valves	S.B.	.13	
					DW-V-156	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	5						
RHR Condensing Mode Steam Supply	21	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	Yes	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-64	MO Gate	O	DC	DC	X	RM	C	C	C	AS-IS	10	16	2						
RCIC Turbine Steam Supply	45	3.2-8 6.2-31e	55	A	RCIC-V-63	MO Gate	I	AC	AC	K	RM	O	O/C	O/C	AS-IS	10	16	-	No	S	Valves	R.B.	No	
					RCIC-V-76	MO Globe	I	AC	AC	K	RM	C	C	C	AS-IS	1	5	-						
					RCIC-V-8	MO Gate	O	DC	DC	X	RM	O	O/C	O/C	AS-IS	4	Std	2						
RCIC Pump Minimum Flow	65	3.2-8 6.2-31h	56	B	RCIC-V-19	MO Globe	O	DC	DC	33	RM	C	C	O/C	AS-IS	2	5	7	No	W	Valves	R.B.	No	22
RCIC Turbine Exhaust	4	3.2-8 6.2-31n	56	B	RCIC-V-68	MO Gate	O	DC	DC	35	~~Manual~~ RM	O	O	O/C	AS-IS	10	Std	10	No	S	Valves	R.B.	No	22
RCIC Turbine Exhaust Vacuum Breaker	116	3.2-8 6.2-31l	56	B	RCIC-V-110	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	9	No	A	Valves	R.B.	No	17
					RCIC-V-113	MO Gate	O	DC	DC	N	RM	O	O	O/C	AS-IS	2	Std	5						
RCIC Vacuum Pump Discharge	64	3.2-8 6.2-31q	56	B	RCIC-V-69	MO Gate	O	DC	DC	36	~~Manual~~ RM	O	O	O/C	AS-IS	1-1/2	Std	4	No	W	Valves	R.B.	No	22
RCIC Pump Suction from Suppression Pool	33	3.2-8 6.2-31n	56	B	RCIC-V-31	MO Gate	O	DC	DC	32	~~Manual~~ RM	C	C	O/C	AS-IS	8	Std	2	No	W	Valves	R.B.	No	23
RPV Head Spray	2	3.2-8 6.2-31e	55	A	RCIC-V-66	Check	I	Process	Process	-	-	C	O	O/C	-	6	-	-	No	W	Valves	R.B.	No	3
					RCIC-V-13	MO Gate	O	DC	DC	34	RM	C	O/C	O/C	AS-IS	6	15	2	No	W	Valves	R.B.	No	
					RHR-V-23	MO Globe	O	DC	DC	L,U, H,R	RM	C	O/C	C	AS-IS	6	Std	7	Yes	W	Valves	R.B.	No	

FSAR Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	Q/C	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
Drywell Spray Loop A	11A	3.2-6 6.2-31g	56	B	RHR-V-16A	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	16	10	26	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17A	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	16	10	24						
Drywell Spray Loop B	11B	3.2-6 6.2-31g	56	B	RHR-V-16B	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	16	10	12	Yes	W	Valves	R.B.	No	17, 24
					RHR-V-17B	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	16	10	2						
LPCI Loop A	12A	3.2-6 6.2-31L	55	A	RHR-V-41A	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42A	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	14	27	21						
LPCI Loop B	12B	3.2-6 6.2-31L	55	A	RHR-V-41B	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42B	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	14	27	20						
LPCI Loop C	12C	3.2-6 6.2-31L	55	A	RHR-V-41C	Check	I	Process	Process	-	-	C	C	O/C	-	14	-	-	Yes	W	Valves	R.B.	No	3, 24
					RHR-V-42C	MO Gate	O	AC	AC	46	RM ~~Manual~~	C	C	O/C	AS-IS	14	27	20						
Shutdown Cooling Return Loop A	19A	3.2-6 6.2-31m	55	A	RHR-V-50A	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123A	MO Gate	I	AC	AC	F,L U,H, R	RM	C	O/C	C	AS-IS	1	STD	-						
					RHR-V-53A	MO Globe	O	AC	AC	H,L, U,R	RM	C	O	C	AS-IS	12	40	5						
Shutdown Cooling Return Loop B	19B	3.2-6 6.2-31m	55	A	RHR-V-50B	Check	I	Process	Process	-	-	C	O	C	-	12	-	-	Yes	W	Valves	R.B.	No	3
					RHR-V-123B	MO Gate	I	AC	AC	F,L U,H, R	RM	C	O/C	C	AS-IS	1	STD	-						
					RHR-V-53B	MO Globe	O	AC	AC	H,L, U,R	RM	C	O	C	AS-IS	12	40	2						
Shutdown Cooling Suction	20	3.2-6 6.2-31k	55	A	RHR-V-9	MO Gate	I	AC	AC	L,U, H,R	RM	C	O	C	AS-IS	20	40	-	Yes	W	Valves	R.B.	No	
					RHR-V-8	MO Gate	O	AC	AC	L,U, H,R	RM	C	O	C	AS-IS	20	40	14						
					RHR-V-209	Check	I	Process	Process	-	-	C	C	C	-	3/4	-	-	No	W	Valves	R.B	No	

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July 1981

FSAR Review

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RHR Loop A:	47	3.2-6 6.2-31p	56	8																				
pump test line					RHR-V-24A	MO Globe	0	AC	AC	F,V	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25A	Relief	0	PP	Spring	-	-	C	C	C	-	2	-	33	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55A	Relief	0	PP	Spring	-	-	C	C	C	-	10	-	22	Yes	S	Valves	R.B.	No	18, 19
heat exch. condensate					RHR-V-11A	MO Gate	0	AC	AC	F,V	RM	C	O/C	C	AS-IS	4	-	18	Yes	W	Valves	R.B.	No	18
heat exch. condensate relief					RHR-RV-36	Relief	0	PP	Spring	-	-	C	C	C	-	8	-	20	Yes	W	Valves	R.B.	No	18, 20
pump minimum flow					RHR-FCV-64A	MO Globe	0	AC	AC	38	RM	C	C	O/C	AS-IS	3	15	22	Yes	W	Valves	R.B.	No	18
heat exch. thermal relief					RHR-RV-1A	Relief	0	PP	Spring	-	-	C	C	C	-	1-1/2	-	188	Yes	W	Valves	R.B.	No	18, 19
heat exch. vent					RHR-V-73A	MO Globe	0	AC	AC	39	RM ~~Manual~~	C	O/C	C	AS-IS	2	Std	175	Yes	A	Valves	R.B.	No	18
FDR system inter-tie					RHR-V-121	Gate	0	Manual	Manual	-	-	LC	LC	LC	-	3	-	6	No	W	Valves	R.B.	No	
CAC system Loop A drain					RHR-V-134A	MO Gate	0	AC	AC	37	RM ~~Manual~~	C	C	O/C	AS-IS	2	Std	44	Yes	W	Valves	R.B.	No	18
pump A suction relief					RHR-RV-88A	Relief	0	PP	Spring	-	-	C	C	C	-	1	-	30	Yes	W	Valves	R.B.	No	18
RHR Loop B	48	3.2-6 6.2-31p	56	8																				
pump test line					RHR-V-24B	MO Globe	0	AC	AC	F,V	RM	C	C	C	AS-IS	18	Std	12	Yes	W	Valves	R.B.	No	2, 18, 24
discharge header relief					RHR-RV-25B	Relief	0	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18, 19
heat exch. steam relief					RHR-RV-55B	Relief	0	PP	Spring	-	-	C	C	C	-	10	-	20	Yes	S	Valves	R.B.	No	18, 19
pump A&B suction relief					RHR-RV-5	Relief	0	PP	Spring	-	-	C	C	C	-	2	-	20	Yes	W	Valves	R.B.	No	18, 19

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November 1980

FSAR Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6).	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RHR Loop A Suppression Pool Suction	35	3.2-6 6.2-31a	56	B	RHR-V-4A	MO Gate	0	AC	AC	46	RM ~~Manual~~	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop B Suppression Pool Suction	32	3.2-6 6.2-31a	56	B	RHR-V-4B	MO Gate	0	AC	AC	46	RM ~~Manual~~	O	O/C	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop C Suppression Pool Suction	36	3.2-6 6.2-31a	56	B	RHR-V-4C	MO Gate	0	AC	AC	46	RM ~~Manual~~	O	O	O	AS-IS	24	Std	2	Yes	W	Valves	R.B.	No	18
RHR Loop A:	117	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RV-95A	Relief	0	PP	Spring	-	-	C	C	C	-	10	-	24	Yes	S	Valves	R.B.	No	18, 19
condensate pot drain					RHR-V-124A	MO Gate	0	AC	AC	39	RM ~~Manual~~	C	C	C	AS-IS	1-1/2	Std	11	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-124B	MO Gate	0	AC	AC	39	RM ~~Manual~~	C	C	C	AS-IS	1-1/2	Std	12	Yes	W	Valves	R.B.	No	18
RHR Loop B:	118	3.2-6 6.2-31d	56	B																				
heat exch. steam relief					RHR-RV-95B	Relief	0	PP	Spring	-	-	C	C	C	-	10	-	21	Yes 19	S	Valves	R.B.	No	18
condensate pot drain					RHR-V-125A	MO Gate	0	AC	AC	39	RM ~~Manual~~	C	C	C	AS-IS	1-1/2	Std	17	Yes	W	Valves	R.B.	No	18
condensate pot drain					RHR-V-125B	MO Gate	0	AC	AC	39	RM ~~Manual~~	C	C	C	AS-IS	1-1/2	Std	14	Yes	W	Valves	R.B.	No	18
RHR Loop C:	26	3.2-6 6.2-31f	56	B																				
pump test line					RHR-V-21	MO Globe	0	AC	AC	F,Y	RM	C	C	C	AS-IS	18	Std	34	Yes	W	Valves	R.B.	No	18
discharge header relief					RHR-RV-25C	Relief	0	PP	Spring	-	-	C	C	C	-	2	-	30	Yes	W	Valves	R.B.	No	18, 19
pump C suction relief					RHR-RV-88C	Relief	0	PP	Spring	-	-	C	C	C	-	1	-	37	Yes	W	Valves	R.B.	No	18, 19
pump minimum flow					RHR-FCV 64C	MO Globe	0	AC	AC	38	RM	O	C	O/C	AS-IS	3	15	30	Yes	W	Valves	R.B.	No	18
Suppression Pool Spray Loop A	25A	3.2-6 6.2-31h	56	B	RHR-V-27A	MO Gate	0	AC	AC	F,Y	RM	C	C	O/C	AS-IS	6	Std	5	Yes	W	Valves	R.B.	No	2, 18, 24
Suppression Pool Spray Loop B	25B	3.2-6 6.2-31h	56	B	RHR-V-27B	MO Gate	0	AC	AC	F,Y	RM	C	C	O/C	AS-IS	6	Std	6	Yes	W	Valves	R.B.	No	2, 18, 24

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November 1980

1 - NK Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
CAC Division 1 discharge to drywell	96	3.2-17 6.2-31g	56	B	CAC-V-2	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	4	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2A	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	6						
CAC Division 2 suction from drywell	97	3.2-17 6.2-31g	56	B	CAC-V-15	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1B	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	4						
CAC Division 2 discharge to drywell	98	3.2-17 6.2-31g	56	B	CAC-V-11	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	8	Yes	A	Valves	R.B.	No	17
					CAC-FCV-2B	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	10						
CAC Division 1 suction from drywell	99	3.2-17 6.2-31g	56	B	CAC-V-6	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	4	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-1A	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 discharge to wetwell	102	3.2-17 6.2-31g	56	B	CAC-V-4	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	3	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4A	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	5						
CAC Division 2 discharge to wetwell	103	3.2-17 6.2-31g	56	B	CAC-V-13	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	7	Yes	A	Valves	R.B.	No	17
					CAC-FCV-4B	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	9						
CAC Division 2 suction from wetwell	104	3.2-17 6.2-31g	56	B	CAC-V-17	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	5	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3B	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	7						
CAC Division 1 suction from wetwell	105	3.2-17 6.2-31g	56	B	CAC-V-8	MO Gate	O	DC	DC	37	RM ~~Manual~~	C	C	O/C	AS-IS	4	Std	2	Yes	A,S	Valves	R.B.	No	17
					CAC-FCV-3A	EHO Globe	O	AC	AC	37	RM ~~Manual~~	C	C	O/C	C	2-1/2	Std	6						

AMENDMENT NO. 3
March 1979

FSAR Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Pent. No.	FSAR Fig. Nos.	GDC	Code Gp. (12)	Valve No.	Valve Type	Loc.	Pwr. to Open (5)	Pwr. to Close (5)	Iso. Sig. (9)	Back Up	Norm Pos. (10)	Shut-down Pos.	Post LOCA	Fail. Pos. (6)	Vlv. Sz. (14)	Close. Time (7) (11)	Dist. to Pent.	Leads to ESF Sys.	Proc. Fld.	Leak Bar. (13)	Term. Zone (13)	Pot. By-pass Leak. (SCFH)	Notes
RCC Inlet Header	5	3.2-14 6.2-31t	56	B	RCC-V-104	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	5	No	W	Valves	R.B.	No	17
					RCC-V-5	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3						
RCC Outlet Header	46	3.2-14 6.2-31o	56	B	RCC-V-21	MO Gate	O	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	3	No	W	Valves	R.B.	No	
					RCC-V-40	MO Gate	I	AC	AC	F,A	-	O	O	C	AS-IS	10	Std	-						
Suppression Pool Cleanup Suction	100	3.2-12 6.2-31l	56	B	FPC-V-153	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	2	No	W	Valves	R.B.	No	17
					FPC-V-154	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	7						
Suppression Pool Cleanup Return	101	3.2-12 6.2-31o	56	B	FPC-V-156	MO Gate	O	AC	AC	F,A	RM	C	C	C	AS-IS	6	Std	3	No	W	Valves	R.B.	No	17
					FPC-V-149	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	6	-	41						
RWCU From Reactor	14	3.2-11 6.2-31k	55	A	RWCU-V-1	MO Gate	I	AC	AC	A,J, E,W	RM	O	O	C	AS-IS	6	Std	-	No	W	Valves	Rad. W.	.35	
					RWCU-V-4	MO Gate	O	DC	DC	A,J, E,Y,W	RM	O	O	C	AS-IS	6	Std	4						
RRC Pump A seal Water	43A	3.2-3 6.2-31c	56	B	RRC-V-13A	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16A	MO Gate	O	AC	AC	45	~~Manual~~ RM	O	O	O	AS-IS	3/4	Std	2						
RRC Pump B seal water	43B	3.2-3 6.2-31c	56	B	RRC-V-13B	Check	I	Process	Process	-	-	O	O	O	-	3/4	Std	-	No	W	Valves	R.B.	No	
					RRC-V-16B	MO Gate	O	AC	AC	45	~~Manual~~ RM	O	O	O	AS-IS	3/4	Std	2						
RRC Sample Line	77Aa	3.2-3 6.2-31d	55	A	RRC-V-19	SO Globe	I	AC	Spring	A,C	RM	C	C	C/O	C	3/4	<5	-	No	W	Valves	T.B.	.05	
					RRC-V-20	AO Globe	O	Air	Spring	A,C	RM	C	C	C/O	C	3/4	Std							

FSAR Revision

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Drywell Equipment Drain	23	3.2-9 6.2-31k	56	B	EDR-V-19	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					EDR-V-20	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	4						
Drywell Floor Drain	24	3.2-10 6.2-31k	56	B	FDR-V-3	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					FDR-V-4	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	3						
Decontamination Soltn. Supply Header	94	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
Decontamination Soltn. Return Header	95	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
CIA for Safety Relief Valve Accumulators	56	3.2-21 6.2-31c	56	B	CIA-V-21	Check	O	Process	Process	-		C	C	C	-	3/4	-	3	No	A	Valves	R.B.	No	17
					CIA-V-20	MO Globe	O	AC	AC	41	RM ~~Manual~~	O	O	O	AS-IS	3/4	Std	10						
CIA Line A for ADS Accumulators	89B	3.2-21 6.2-31c	56	B	CIA-V-31A	Check	O	Process	Process	-		C	C	C	-	1/2	-	5	No	A	Valves	R.B.	No	17
					CIA-V-30A	MO Globe	O	AC	AC	42	RM ~~Manual~~	O	O	O	AS-IS	1/2	Std	15						
CIA Line B for ADS Accumulators	91	3.2-21 6.2-31c	56	B	CIA-V-31B	Check	O	Process	Process	-		C	C	C	-	1/2	-	2	No	A	Valves	R.B.	No	17
					CIA-V-30B	MO Globe	O	AC	AC	42	RM ~~Manual~~	O	O	O	AS-IS	1/2	Std	15						
CRD Insert Lines (185 separate lines)	9	3.2-4	56	B						See Note 4														
CRD Withdrawal lines (185 separate lines)	10	3.2-4	56	B						See Note 4														

CONTAINMENT SYSTEMS BRANCH

ISSUE 19

NRC:	Page 6.2-124, LPCI with a postulated break in the system after a LOCA-this line has no signals provided to indicate the break.
Supply System:	Passive failure post-LOCA is not a guillotine break but a seal failure. Sump level alarms will provide indication of a break.
Summation:	F. Eltawila will review LaSalle FSAR and inform the Supply System of requirements.
	See Item 24.

ISSUE 24

NRC: See LaSalle LPCI injection line. Note 31 in LaSalle FSAR, Table 6.2-21. penetrations M-12 through M-14 have remote manual and other signals to isolate these lines. Identify these signals and evaluate applicability to WNP-2.

Supply System: LaSalle will be contacted and an evaluation as to applicability to WNP-2 will be made by October 2, 1981 (see issue 19).

Supplementary Information: The WNP-2 LaSalle isolation valve designs for the LPCI injection valve are similar. The only difference is that LaSalle has a 3/4 inch air-operated globe valve in parallel with the check valve inside containment, which have isolation signals to close. WNP-2 has no such bypass line. Neither WNP-2 or LaSalle has isolation signals to close the LPCI injection valves. WNP-2 uses valve packing leakage indication, and reactor bldg. sump level indication, as well as reactor water level and suppression pool water level instrumentation to alert the operator of excessive leakage.

Issues 19 and 24

The LaSalle isolation design for the LPCI containment penetration is similar to the WNP-2 design in that containment isolation is provided by a motor operated gate valve outside containment and a testable check valve inside containment, (see Figure 1). The only difference in the design is that LaSalle has a 3/4-inch globe valve in parallel with the testable check valve inside containment.

The 3/4" air operated globe valve receives the reactor vessel low-low water level and high drywell isolation signals, has remote manual operation from the control room and fails closed on loss of power. The only signal to the LPCI injection line isolation valves, however, is a remote manual signal to the gate valve outside containment, which is identical to the isolation valve control logic at WNP-2.

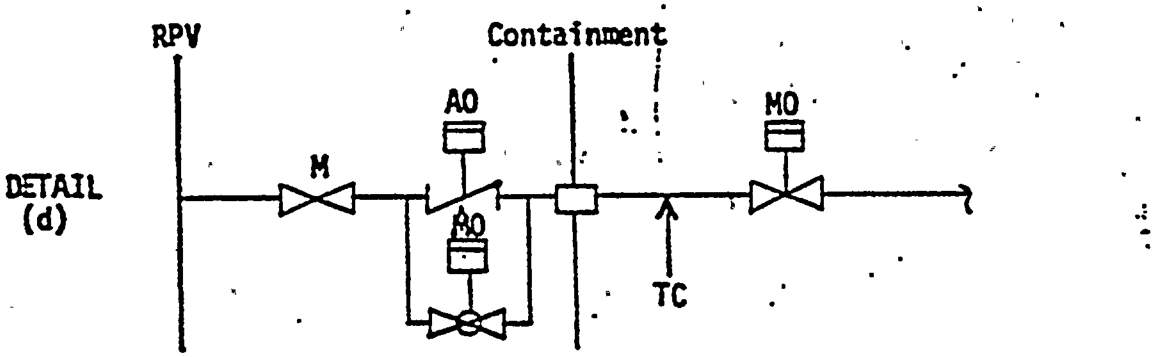
LaSalle and WNP-2, therefore, both depend on:

1. Leak detection systems consisting of sumps with redundant level indication in the control room and stem leak off instrumentation to determine the location of a leak or line failure and allow the operator to close the isolation valve associated with the line.

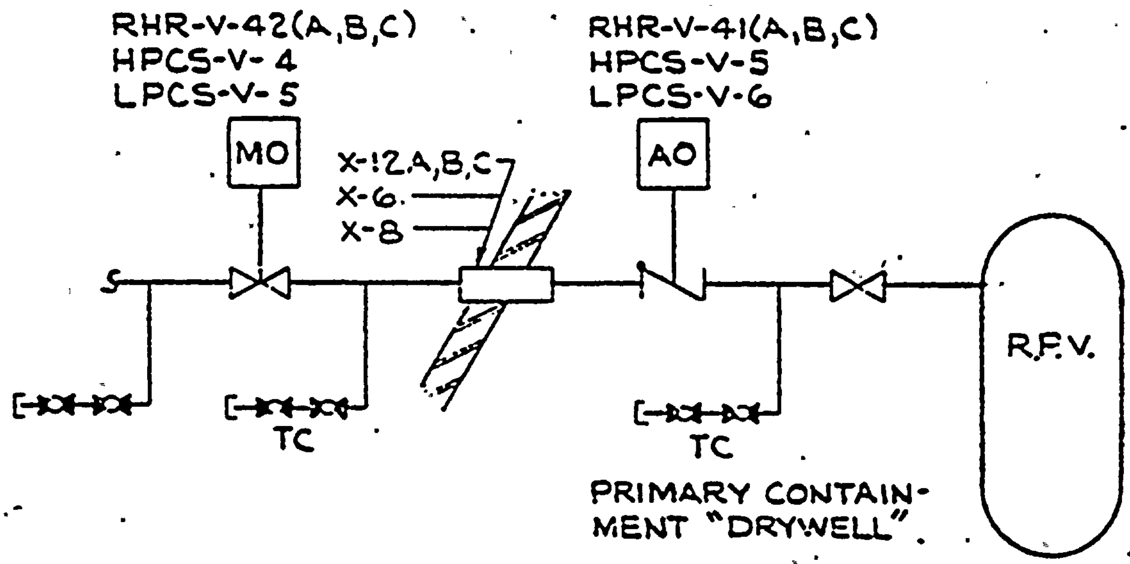
2. RPV level information to ascertain whether or not the flow is actually reaching the RPV.

3. Suppression pool water level information to identify the occurrence of leakage or a line failure.

FIGURE



LA SALLE. ISOLATION VALVE ARRANGEMENT FOR LPCI LINE



WNP-2 ISOLATION VALVE ARRANGEMENT FOR LPCI LINE

CONTAINMENT SYSTEMS BRANCH

ISSUE 20

NRC: Page 6.2-131, penetrations 56, 89b, 91 and 82E show check valves outside containment. Check valves outside containment are not acceptable.

Supply System: The Supply System will relocate check valves and containment isolation valves inside containment. See revised FSAR pages 6.2-130 and 6.2-131 and Figures 6.2-31c and 6.2-31r (attached).

TABLE 6.2-16 (Continued)

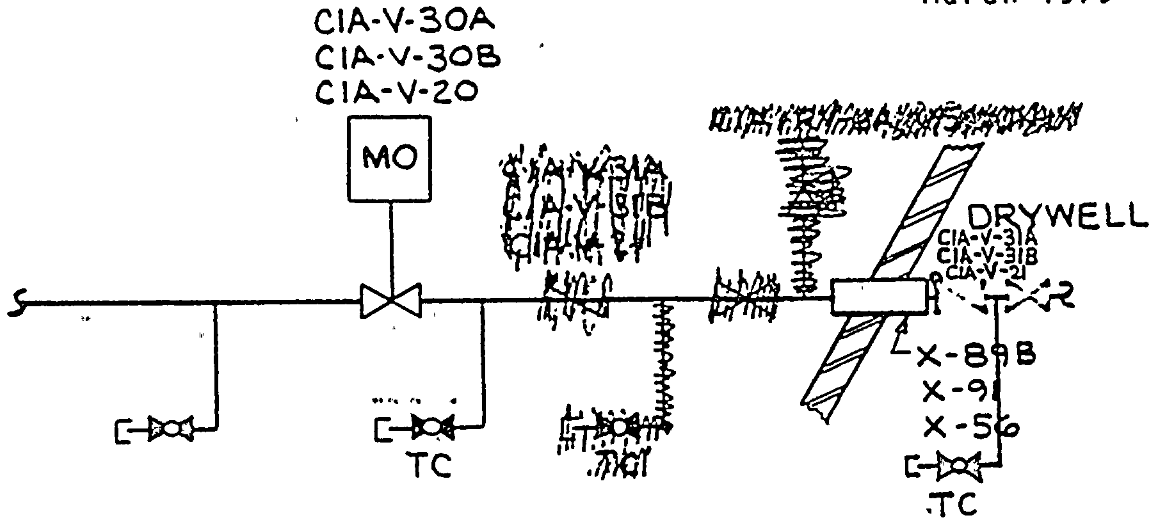
LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp. (12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Drywell Equipment Drain	23	3.2-9 6.2-31k	56	B	EDR-V-19	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					EDR-V-20	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	4						
Drywell Floor Drain	24	3.2-10 6.2-31k	56	B	FDR-V-3	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	2	No	W	Valves	R.B.	No	17
					FDR-V-4	AO Gate	O	Air	Spring	F,A	RM	O	O	C	C	3	Std	3						
Decontamination Soltn. Supply Header	94	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
Decontamination Soltn. Return Header	95	3.2-10	NA	B	-	-	-	-	-	-	-	-	-	-	-	4	-	-	-	-	Blanked Close	R.B.	No	
CIA for Safety Relief Valve Accumulators	56	3.2-21 6.2-31c	56	B	CIA-V-21	Check	I	Process	Process	-	-	C	C	C	-	3/4	-	-	No	A	Valves	R.B.	No	[illegible]
					CIA-V-20	MO Globe	O	AC	AC	41	Manual	O	O	O	AS-IS	3/4	Std	10						
CIA Line A for ADS Accumulators	89B	3.2-21 6.2-31c	56	B	CIA-V-31A	Check	I	Process	Process	-	-	C	C	C	-	1/2	-	-	No	A	Valves	R.B.	No	[illegible]
					CIA-V-30A	MO Globe	O	AC	AC	42	Manual	O	O	O	AS-IS	1/2	Std	15						
CIA Line B for ADS Accumulators	91	3.2-21 6.2-31c	56	B	CIA-V-31B	Check	I	Process	Process	-	-	C	C	C	-	1/2	-	-	No	A	Valves	R.B.	No	[illegible]
					CIA-V-30B	MO Globe	O	AC	AC	42	Manual	O	O	O	AS-IS	1/2	Std	15						
CRD Insert Lines (185 separate lines)	9	3.2-4	56	B						See Note 4														
CRD Withdrawal lines (185 separate lines)	10	3.2-4	56	B						See Note 4														

AMENDMENT NO. 5
August 1979

TABLE 6.2-16 (Continued)

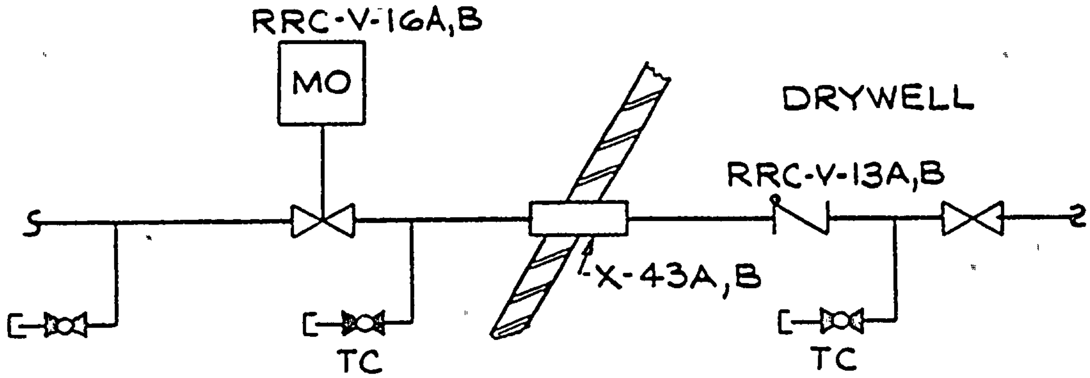
LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Air line for testing RHR-V-50A	42d	6.2-31r 3.2-6	56	B	PI-VX-42d	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-216	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-50B	69c	6.2-31r 3.2-6	56	B	PI-VX-69c	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-221	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41A	61f	6.2-31r 3.2-6	56	B	PI-VX-61f	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-219	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41B	54Bf	6.2-31r 3.2-6	56	B	PI-VX-54Bf	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-218	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41C	62f	6.2-31r 3.2-6	56	B	PI-VX-62f	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-220	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing LPCS-V-6	78d	6.2-31r 3.2-7	56	B	PI-VX-78d	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-222	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing HPCS-V-5	78e	6.2-31r 3.2-7	56	B	PI-VX-78e	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	valves	R.B.	No	25
					PI-VX-223	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RCIC-V-66	54Aa	6.2-31r 3.2-8	56	B	PI-VX-54Aa	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-217	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing WW-DW vacuum relief valves	82e	6.2-31r 9.3-1	56	B	CAS-V-453	SO Globe	O	AC	Spring	[illegible]C,F	-	C	C	C	C	1	<5	5	No	A	Valves	R.B.	No	[illegible]34
					CAS-CVX-82e	Check	I	Process	Process	-	-	C	C	C	-	1	-	[illegible]-						
Air line for maintenance	93	9.3-1 6.2-31t	56	B	-	Pipe Cap	I	-	-	-	-	C	C	C	-	2	-	-	No	A	Cap & Valve	S.B.	No	
					SA-V-109	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	1						
Tip lines	27a-e		54	-	C51J004	SO Ball	O	AC	AC	L,F	RM	C	C	C	C	3/8	<5	2	No	A	Valves	R.B.	No	29
					C51J004	Shear	O	-	Explosive	43	-	O	O	O	O	3/8	-	2						

AMENDMENT NO. 5
August 1979



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

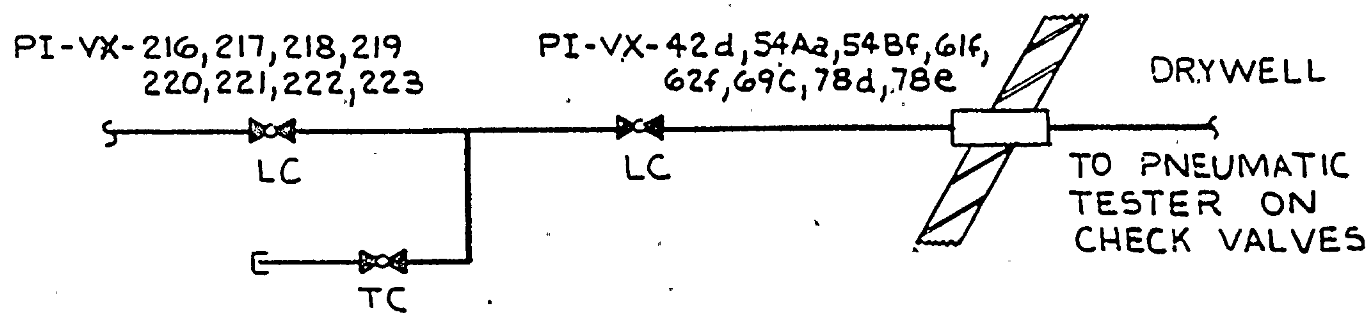
CONTAINMENT INSTRUMENT AIR



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

RRC PUMP SEAL PURGE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION VALVE ARRANGEMENT FOR PENETRATIONS - X-89B, X-91, X-56 & X-43A, X-43B	FIGURE 6.2-31c



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

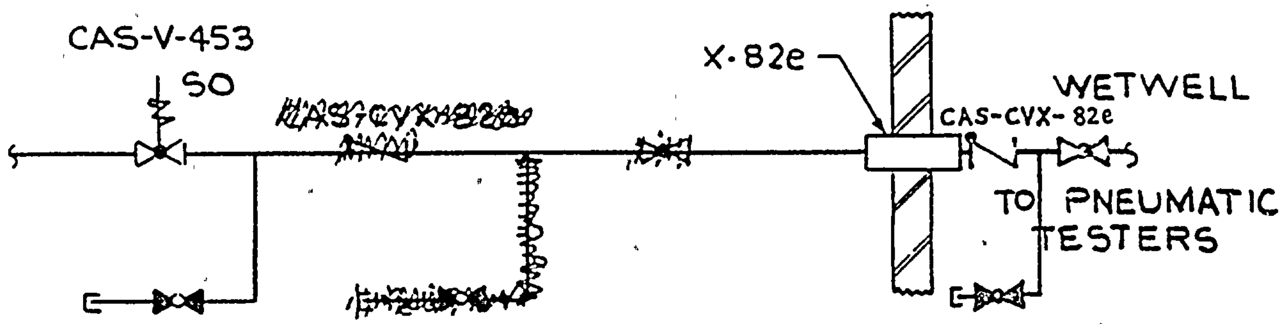
X-61f AIR LINE FOR TESTING RHR-V-41A

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

Amendment No. 3
March 1979

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

CONTAINMENT SYSTEMS BRANCH

ISSUE 23

NRC:	Paragraph 6.2.4.3.2.2.1.3 - Please clarify the wording of this subsection, and add more detail to Figure 6.2-31p.
Supply System:	Figure 6.2-31p will be revised, and the text clarified in the FSAR. See revised FSAR page 6.2-64 and Figure 6.2-31p (attached).

6.2.4.3.2.2.1.3 RHR Heat Exchanger Vent Lines

replace with attached

~~The RHR heat exchanger vent lines discharge to the suppression chamber via relief valve discharge lines and are provided with two remotely controlled motor operated globe valves. Connections to the relief discharge lines are isolated by the relief valves themselves in a fashion similar to a check valve. The addition of block valves for isolation would therefore defeat the purpose for which the relief valves are installed.~~

6.2.4.3.2.2.1.4 RHR Relief Valve Discharge Lines

The RHR relief valve discharge to the suppression pool has no valve other than the relief valve. This relief valve will not be opened during normal operation and, therefore, can be considered as normally closed and adequate under the same criteria as the suppression chamber spray line explained in 6.2.4.3.2.2.3.4.

6.2.4.3.2.2.2 Effluent Lines from Suppression Chamber

The RHR, RCIC, LPCS, and HPCS suction lines contain motor-operated, remote manually actuated, gate valves which provide assurance of isolating these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression chamber is considered an extension of containment since it must be available for long-term usage following a design basis loss-of-coolant accident, and as such, is designed to the same quality standards as the containment. Thus, the need for isolation is conditional. The ECCS discharge line fill system (ECCS waterleg pumps) takes suction from the respective ECCS pump effluent line from the suppression pool downstream of the isolation valve. The ECCS discharge line fill system suction line has a manual valve for operational purposes. This system is isolated from the containment by the respective ECCS pump suction valve from suppression pool as listed in Table 6.2-16.

6.2.4.3.2.2.3 Influent and Effluent Lines from Drywell and Suppression Pool Free Volume

6.2.4.3.2.2.3.1 Containment Atmosphere Control Lines

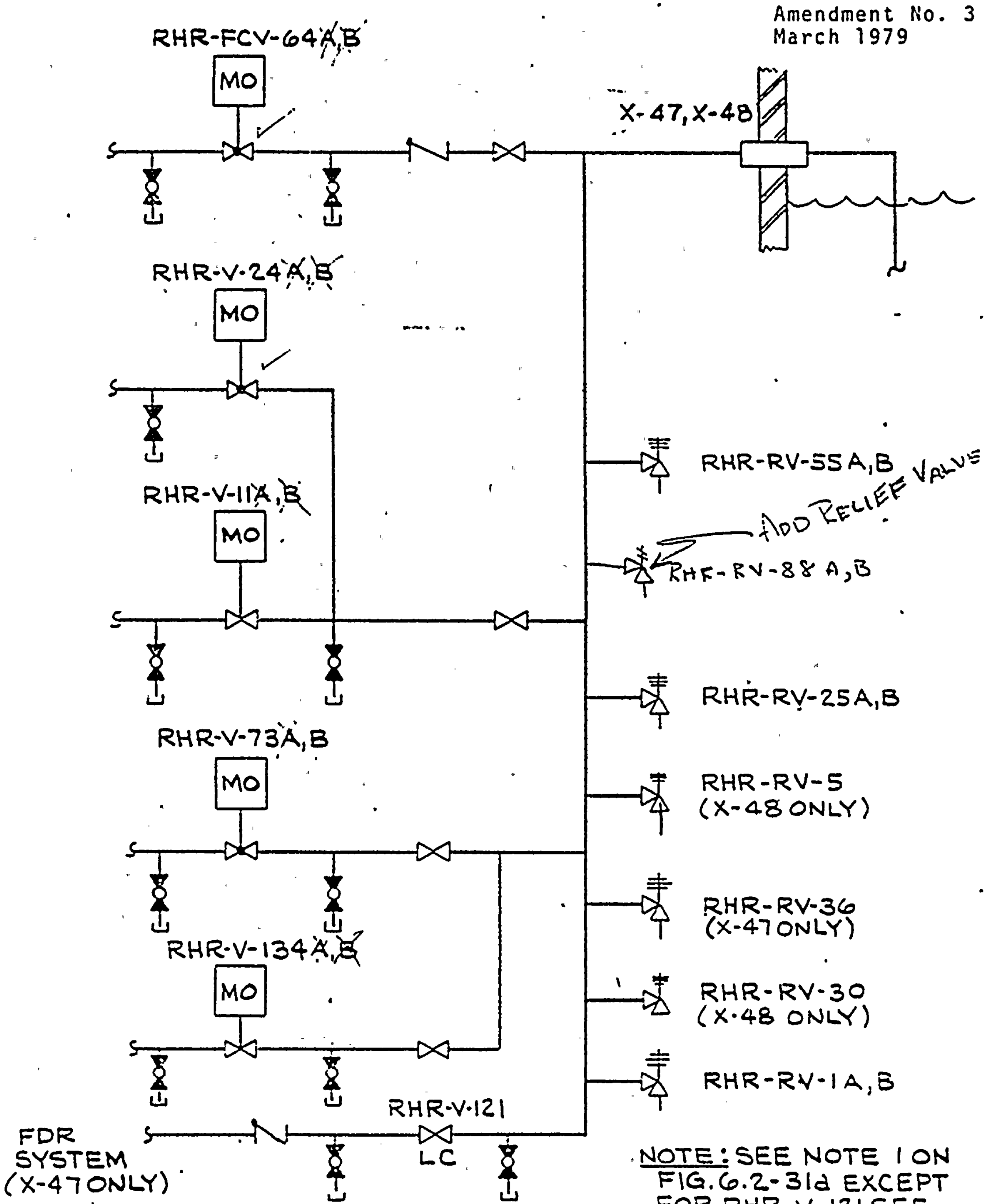
The containment atmosphere control system lines which penetrate the containment are equipped with two power-operated valves in series, normally closed, remote manually actuated

Issue 23

6.2.4.3.2.2.1.3 RHR Heat Exchanger Vent Lines:

The RHR heat exchanger vent lines discharge through the RHR heat exchanger relief valve discharge lines and the RHR loop A and loop B test lines to the suppression pool. Two globe valves in each vent line provide the system pressure boundary and are used to control venting during the RHR heat exchanger filling and draining operations. The outboard globe valve in each line is also considered as, and meets the criteria for, a containment system isolation valve. Both valves are normally closed remotely controlled motor operated globe valves. Each vent line is also equipped with a manual block valve and the test connections necessary for Type C testing of the isolation valve.

Amendment No. 3
March 1979



NOTE: SEE NOTE 1 ON FIG. 6.2-31d EXCEPT FOR RHR-V-121 SEE NOTE 5. See Note 8 For Testing Of Test Connection

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 6.2-31f

CONTAINMENT SYSTEMS BRANCH

ISSUE 25

NRC: Perform a pre-op ILRT at full pressure for twenty-four (24) hours with all subsequent tests at a twelve (12) hour duration.

Supply System: The Supply System will provide justification for performance of the test at a twelve (12) hour duration at the ILRT frequency by October 2, 1981.

Supplementary Information: Justification for performance of the subsequent ILRT tests over a 12-hour duration, instead of a 24-hour duration, will be provided by November 6, 1981. See revised FSAR pages 6.2-81 and 6.2-83 (attached).

CONTAINMENT SYSTEMS BRANCH

ISSUE 29

NRC: Question 031.070, containment system drywell/wetwell leak test. The FSAR should be changes to indicate that the drywell/wetwell leak test be performed at a higher pressure than 1 psi.

The pressure for the test should be approximately submergence pressure, 5 psig.

Supply System: The Supply System proposes to perform tests at the following frequency:

a. Pre-operational Test: $1\frac{1}{2}$ & 5 psig
b. First Refueling: $1\frac{1}{2}$ psig
c. First ILRT Interval: $1\frac{1}{2}$ & 5 psig
d. WNP-2 will demonstrate that $1\frac{1}{2}$ psig test is adequate, based on information obtained from items a-c above, and will request discontinuation of the 5 psig and perform $1\frac{1}{2}$ psig tests at each succeeding refueling outage.

The basis for the 5 psi test is that a 5 psi differential pressure is relatively high but does not clear the water column out of the downcomers. The $1\frac{1}{2}$ psi test will be demonstrated to be adequate by comparison with a 5 psi test. The advantage of a $1\frac{1}{2}$ psi test is that tests can be performed either just prior to a shutdown or during a startup without causing a containment isolation or scram.

Summation: F. Eltawila will review the Supply System position and advise the Supply System of acceptability.

6.2.6 CONTAINMENT LEAKAGE TESTING

As described below, General Design Criteria 52, 53 and 54 have been met.

6.2.6.1 Containment Integrated Leakage Rate Test (Type A Tests)

The WNP-2 primary containment system is a steel pressure suppression system of the over and under configuration with a designed leakage rate of 0.5 percent by volume per day at 45 psig. A maximum allowable integrated vessel leak rate of 0.5 percent by weight per day at 34.7 psig has been established to limit leakage during and following the postulated DBA to less than that which would result in offsite doses greater than those specified in 10CFR Part 100. Leakage rate tests at reduced pressures may be established such that the measured leakage rate does not exceed the maximum allowable at that reduced pressure.

A structural integrity test (SIT) involving pneumatic pressurization of the drywell and suppression chamber was performed at 51.8 psig, 1.15 times the containment vessel design pressure of 45 psig. This test was conducted in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article NE-63000 (1971). This test is described in detail in 3.8.2.7.

Preoperational testing shall involve performing a Type A integrated leakage rate test. This test and the periodic tests required during plant operation will be conducted in accordance with 10CFR50 Appendix J, Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors. The preoperational test will involve the following sequence and steps:

a. Preparation of containment vessel, test instrumentation and test equipment.

~~b. Performing a reduced pressure test at not less than 17.4 psig.~~

b ~~c~~. Performing a peak pressure test at not less than 34.7 psig.

The periodic tests will be performed at the peak pressure. ~~or at a reduced pressure not less than 17.4 psig as established during preoperational testing.~~

d. Control rod drive system

e. Low pressure core spray system

f. Standby liquid control system

g. Reactor water cleanup system

h. Feedwater system

The reactor closed cooling system is a closed system which penetrates containment and returns without being open to containment atmosphere. This system will not be vented because the system is required to maintain containment temperature during the ILRT. The control rod drive, standby liquid control, reactor water cleanup, and reactor feedwater systems are required to maintain the plant in a safe shutdown and will not be vented. The residual heat removal, high pressure core spray, low pressure core spray and reactor core isolation cooling systems will not be vented since they will be filled with water and operating during the post accident conditions. The systems which will not be vented are designed with isolation valves which will undergo Type C tests.

The containment integrated leak rate tests will be scheduled, to the extent practicable, during a period of forecasted constant meteorological conditions. Prior to the start of the test, containment test conditions, temperature, pressure and humidity will be monitored for a period of about four hours to ensure stabilization of containment conditions.

The test methods, procedures, test equipment, facilities, testing period, means of local leakage testing, and test leak rate accuracy verification will be in accordance with ANSI N 45.4-1972.

Acceptance criteria for the preoperational and periodic leak rate tests shall be based on the criteria given in 10CFR50, Appendix J. Peak pressure test performed at the calculated peak containment pressure, P_a (34.7 psig) will be acceptable provided the total measured containment leakage rate L_{am} does not exceed 75% of the maximum allowable leakage rate, L_a. ~~Reduced pressure tests (not less than 17.4 psig) will be acceptable provided the total measured containment leakage rate,~~ $\cancel{L_{tm}}$, ~~does not exceed 75% of the maximum allowable leakage rate~~ $\cancel{L_t}$ ~~determined at the reduced pressure,~~ $\cancel{P_t}$, ~~during preoperational testing.~~ If following periodic tests, local leakage tests are performed so that repairs can be accomplished to reduce the total leakage within the acceptance

Insert attached

Insert

The initial preoperational type A test will be performed for a duration of twenty-four (24) hours at P_a. Subsequent tests will be performed for a minimum of 12 hours and will be continued until the 95 percent upper confidence limit is less than or equal to 0.75 L_a.

CONTAINMENT SYSTEMS BRANCH

ISSUE 26

NRC:	When secondary containment pressurization test is conducted, what pressure is used in secondary containment to verify draw-down time of the stand-by gas treatment system? After a LOCA, the pressure in secondary containment will be positive. What relationship will this test have to the post-LOCA secondary containment pressure? Will the test bound the analysis? Provide Figure 6.2-42 and 6.2-43.
Supply System:	The pressure for the secondary containment pressurization test and it's relationship to a post-LOCA pressure in the secondary containment was presented and the WNP-2 position is acceptable.

Pages 6.2-50a and 6.2-107 of the WNP-2 FSAR will be revised to indicate that the referenced Figures 6.2-42 and 6.2-43 have been replaced by Table 6.2-29.

Refer to NRC Question 022.071.

See revised FSAR pages 6.2-50a and 6.2-107 (attached).

Studies were performed to determine the temperature and pressure response of the Reactor Building following a loss-of-coolant accident (LOCA). The results of this analysis indicate that the building lighting load and the spent fuel decay heat are the two dominant heat loads.

Acceptable Reactor Building temperature and pressure is maintained by extinguishing all normal lighting automatically by isolation signal and keeping one fuel pool cooling loop operating following a LOCA. Emergency lighting will be on for one hour following a LOCA to facilitate the evacuation of the Reactor Building.

6.2.3.3.1.1 Summary and Conclusions

The post-LOCA transient response of the secondary containment atmosphere has been analyzed for a duration of 200 hours. The characteristics of the transient responses may be summarized as follows:

a. The postulated chronological sequence of events in secondary containment during post-LOCA period are tabulated in Table 6.2-28.

b. The transient response of pool water temperature, reactor building air temperature and exhausted air volume (in CFM) are ~~plotted in Figure 6.2-42.~~ shown in Table 6.2-29.

c. The post-LOCA transients of the secondary containment are tabulated as shown in Table 6.2-29 in terms of: (a) building air temperature; (b) building air humidity ratio; (c) spent fuel pool water temperature; (d) evaporation rate from the spent fuel pool; (e) structure steel temperature; (f) building pressure; (g) exhaust air volume rate required to maintain the building pressure at negative pressure of 1/4-inch water gage.

d. The short-term post-LOCA secondary containment pressure transient is shown in ~~Figure 6.2-43.~~ Table ~~6.2~~ 6.2-29. The secondary containment pressure equalizes with the environment 5 seconds after a LOCA, then continues to build up until the standby gas treatment fan starts at 34 seconds after LOCA. The secondary containment air pressure continues to decrease until it reaches a negative pressure of 1/4 inch w.g. at the 150 seconds after a LOCA.

TABLE 6.2-12

SECONDARY CONTAINMENT DESIGN AND PERFORMANCE DATA

I. Secondary Containment Design

A. Free Volume: 3.5×10^6 ft^3; the entire secondary containment is considered as one volume.

B. Pressure

1. Normal Operation: -0.25" water gauge (with respect to the outside atmosphere)

2. Post accident: -0.25" water gauge

C. Infiltration rate during post accident period: 100% of free volume in a 24-hour period.

D. Exhaust Fans (Standby Gas Treatment System): Two independent and redundant filter trains each with two 100% exhaust fans (see 6.5.1)

II. Transient Analysis

A. The transient on the secondary containment after a design basis LOCA is evaluated in 6.2.3.3.1. The temperature and pressure transient is summarized in ~~Figure 6.2-42, and~~ Table 6-2-29.

B. Thermal Characteristics

1. Primary Containment Wall

	Thickness	Thermal Conductivity BTU/hr ft -°F	Thermal Capacitance BTU/ft^3 - °F
Steel liner	1.5"	30	52.5
Polyurethane	2.25"	.022	1.2
Fiberglass	.31"	.022	1.6
Concrete	5.5'	.64	29.8

CONTAINMENT SYSTEMS BRANCH

ISSUE 30

NRC: Page 6.2-32 - Purging during normal operations - statement requires revision due to containment inerting commitment.

Supply System: The FSAR will be revised, and changes made, to reflect inerted conditions when the FSAR text is added for containment inerting. The rewrite will show purging for inerting, deinerting, and pressure control due to pneumatic leakage. Purging will not be done for temperature or humidity control. The text revision will eliminate the 1% lifetime limit for the purge valves.

See attached FSAR pages 6.2-31, 32, and 33.

6.2.1.1.8 Primary Containment Environmental Control

6.2.1.8.1 Temperature, Humidity, and Pressure Control During Reactor Operation

The drywell is maintained at its normal operating ~~limit~~ TEMPERATURE RANGE of 135°F - 150°F by the use of any 2 of the 3 lower containment coolers and either one of the two upper containment coolers mounted in the drywell area. The cooling coils for these units are supplied with water at 95°F, or less, from the reactor building closed cooling water system. Six of the nine recirculating fans mounted at various locations in the drywell aid in circulating the drywell ~~air~~ ATMOSPHERE. The remaining two unit coolers and three recirculating fans are held in standby (see Figure 9.4-8).

Since there is no heat producing equipment in the wetwell, it can be maintained below 95°F without air cooling equipment.

The unit coolers are sufficient to control the temperature and humidity from all expected heat sources and leaks during normal reactor operation. The containment purge system will not be used to control containment temperature or humidity during reactor operation.

To relieve pressure during reactor operation, the operator can establish a flow path from the drywell to the reactor building exhaust system or to the standby gas treatment system through the drywell purge exhaust line described in 6.2.1.1.8.2. By opening the 2" bypass valves around ~~one of the~~ THE ~~two~~ purge exhaust valves rather than the ~~MS~~ purge exhaust valve itself, the operator can limit the flow to 170 scfm. This flow is adequate for a drywell atmosphere temperature rise from 70°F to 150°F in three hours while maintaining the primary containment at no greater than .5 psi above the reactor building pressure. The 2" bypass valves would limit the RADIOactivity released prior to valve closure to a very small amount in the unlikely event a LOCA occurs with the vent path open. If necessary, the wetwell can be vented in a similar way to relieve pressure.

The reactor building-to-wetwell and wetwell-to-drywell vacuum breakers operate automatically to control containment vacuum.

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 AND DE-INERT THE CONTAINMENT 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 OF NITROGEN once a year during scheduled refueling shutdown period and as required for inspection OR MAINTENANCE. 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. THE Purged ~~air~~ NITROGEN 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 9.4-8 and 9.4-2). If a high airborne activity occurs, the radiation monitors at the exhaust air plenum would cause the reactor building's ventilation and primary containment purge systems to isolate.

Provision is also made to purge NITROGEN FROM THE 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. THE Purged ~~air~~ NITROGEN 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 WILL be used during reactor operation only for purging the primary containment prior to personnel entry.~~ Purge system operation during reactor operation including startup, hot standby, and hot shutdown will be limited ~~to less than 1% of reactor operating time~~ TO INERTING, THROUGH THE PURGE SYSTEM, DE-INERTING AND PRESSURE CONTROL. THE PURGE SYSTEM WILL NOT BE USED FOR TEMPERATURE OR HUMIDITY CONTROL.

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, 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. INSERT.

6.2.1.1.8.3 Post - LOCA

CONTINUED CONTAINMENT INERTING IS NOT REQUIRED AFTER A LOCA SINCE

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. Two 100% redundant hydrogen recombiners are available to be placed in operation to ~~ensure that~~ ENSURE THAT the hydrogen/OXYGEN buildup does not reach a flammable level. THE Containment purge SYSTEM has the capability for a controlled purge of the containment atmosphere to aid in hydrogen control, if necessary.

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.

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'.

Insert (to Page 6.2-33):

Dual isolation valves are also provided on the nitrogen inerting makeup piping connecting to the purge piping downstream of the 30" and 24" isolation valves. This permits up to 75 cfm of nitrogen to be added to the containment during reactor operation to compensate for the postulated leakage listed in Table 6.2-1.

CONTAINMENT SYSTEMS BRANCH

ISSUE 31

NRC:	Page 6.2-4 - Blown-off panels indicated as existing in primary containment. Please clarify.
Supply System:	The FSAR text will be revised to reflect insulation panels that are blown-off.
	See revised FSAR page 6.2-4 (attached).

environmental controls during a LOCA. All equipment required to mitigate the consequences of an accident is designed to perform the required functions for the required duration of time in the accident environment. The equipment accident environment is listed in Table 3.11-2.

Reflective metal insulation, manufactured and installed in panels, is used exclusively within the primary containment.

The panels used for the pipes are typically 2 feet long, 3" - 4" thick, and cover half of the pipe's circumference. These panels have 24 gauge stainless steel sheets which fully encase the 6 mil aluminum sheets. The panels used for the RPV are larger, typically 2' x 6', and are encased by 18 gauge stainless steel.

All panels on piping covering areas which require inservice-inspection, such as welds, are fastened by quick release buckle bands. Non-removable insulation panels around pipes are fastened, one to another, using self taping screws.

The fasteners have been designed to be weaker than the panels; and therefore, it is postulated that some panels near a pipe break will be blown away but that the panels themselves will not be sheared open.

The ~~blown off~~ insulation panels that have blown off constitute the only credible debris within the primary containment following a LOCA and seismic event. All equipment within the primary containment, if not designed to Seismic I standards, is at least supported so as to remain fastened during a seismic event.

Large pieces of debris are not considered to have deleterious effects on the containment systems. The grating (see Figure 6.2-24) at the 501'-0" elevation, which covers approximately 80% of primary containment cross sectional area, would stop the majority of the loose insulation panels. Any of the remaining panels could be pressed against the outer perimeter of the jet deflectors, but it is not considered credible that the panel could enter the actual downcomer vent. Partial blockage of several jet deflectors would have an insignificant effect on the containment vent system.

CONTAINMENT SYSTEMS BRANCH

ISSUE 32

NRC:	Airlock doors. Commit to testing airlock doors within 72 hours of last closing.
Supply System:	The Supply System will commit to testing airlock doors within 72 hours of last closing.
	Attached Revised Section 6.2.6.4 incorporates Appendix J Revisions and reflects FSAR Tech. Spec 3/4.6.1.3 Position.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The preoperational Type A, B and C leakage tests will be completed prior to any reactor operating period. After the preoperational leakage testing, periodic tests shall be performed in accordance with the following schedule:

a. Containment Integrated Leakage Rate Test (Type A)

A set of three Type A tests shall be performed at approximately equal intervals during each ten year service period. The third test of each set shall be conducted when the plant is shut down for the ten year plant service inspection.

b. Containment Penetration Leakage Rate Test (Type B)

Type B tests shall be performed during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. ~~Air locks shall be tested at six month intervals and following each opening during such intervals, the air lock door seals shall be tested by pressurization of the space between the seals of each door.~~ During Periods When Containment Integrity Is Required, Air Locks Shall Be Tested At Six Month Intervals And Air Lock Door Seals Tested Within 72 Hours Of Each Air Lock Opening (Or At Least Once Every 72 Hours During Multiple Openings). When Air Lock Doors Have Not Been Opened During The Operating Period, The Six Month Leakage Test Can Be Deferred And Performed During Reactor Refueling Outage Or Other Convenient Intervals But In No Case At Intervals Greater Than 18 Months. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests can be performed every other refueling outage, but in no case at intervals greater than three years.

c. Containment Isolation Valve Leakage (Type C)

Type C tests shall be performed during each refueling outage, but in no case at intervals greater than two years.

Results of the preoperational and subsequent periodic tests shall be summarized in a technical report submitted to the commission approximately three months after the conducting of each test. These reports shall be prepared in accordance with the requirements of 10CFR50, Appendix J. Leakage tests which failed to meet their respective acceptance criteria shall be reported in a separate accompanying summary report that includes an analysis and interpretation of the test data.

CONTAINMENT SYSTEMS BRANCH

ISSUE 33

NRC: Where in the FSAR is mass/energy subcooling based on General Electric methodology (NEDO-10320)?

Supply System: The containment pressure response presented in the FSAR did not consider subcooling as presented in NEDO-10320. The peak drywell pressure occurs at about 18 seconds and subcooling has little effect on the peak pressure.

The analysis for annulus pressurization did include the effect of subcooling.

The calculation of pool swell that was performed included the effects of subcooling and pipe inventory. This is presented in the WNP-2 DAR.

See revised response to Q. 222.002 and revised FSAR pages 6.2-33 through 6.2.33e (attached).

Q. 222.002

Provide a detailed description of your analytical model to evaluate the mass and energy release rates for your analyses of the short-term annulus pressurization and the evaluation of the structural loads resulting from postulated pipe breaks for the first five seconds following the accident. Indicate the mass flux (LB_M/sec-ft^2), the enthalpy (BTU/LB_M) and the flow area (square feet) as a function of time for each side of the break. Justify all your assumptions. Describe the break geometry assumed throughout the transient. Discuss the overall conservatism of your analysis.

Response:

Extensive documentation has been submitted by WPPSS to NRC concerning mass and energy release rates for short-term annulus pressurization in response to a post-construction permit item on the sacrificial shield design. Please refer to references 3.8-5, 3.8-6, and 3.8-7 of the FSAR for the requested information (referenced from 3.8.3.1.2 and revised 6.2.1.2). Copies of these references have been submitted to the NRC before and more recently to Mr. Jack Kudrick of Containment Systems Branch via Reference 1. The NRC in References 2 and 3 found the WPPSS reports acceptable. ~~The information may be updated in the future, however, to reflect changes necessitated by current piping system analysis. As such, these revisions will be reflected in amendments to 6.2.1.2 of the FSAR.~~

In summary, though, for the short-term annulus pressurization analysis and subsequent evaluation of structural loads the analyltical model to evaluate mass and energy release rates is the simple and conservative Moody's two phase critical flows model. ~~For a break in LPCI, HPCS, LPCS, feedwater, and recirculation lines, the following assumptions are used:~~
The assumptions for the RFW and RRC lines analysis is described in revised 6.2.1.2

~~a. The break is the double-ended guillotine type which opens instantaneously.~~

~~b. For the time of interest (the first few seconds of the break) the blowdown flow rate and energy are constant.~~

~~c. Water from the reactor side of the break is saturated water at 1060 psia and enthalpy of 550 Btu/lb.~~

~~d. Moody's critical flows rate is 8100 lb_m/sec/ft^2 for the conditions listed in c.~~

e. Reactor depressurization or change in flow quality is not considered.

f. Initial fluid inventory in the pipe is depleted at a rate consistent with the enthalpy of the fluid in the pipe.

g. Frictional loss of flow in the pipe is neglected.

The constant flow approach is used because it is more conservative than the time dependent blowdown calculation. Assumptions a, e and g are also intended to maximize the break flow rate, and therefore to produce conservative results. For additional conservatism, the enthalpy of flow from the pipe side is taken to be the same as the enthalpy of flow from the reactor side (550 Btu/lb).

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, 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. Two 100% redundant hydrogen recombiners are available to be placed in operation to ensure that the hydrogen buildup does not reach a flammable level. Containment purge has the capability for a controlled purge of the containment atmosphere to aid in hydrogen control, if necessary.

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.

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'.

Subcompartment analyses for a postulated high energy pipe break in the primary containment were performed for the annulus inside the sacrificial shield wall, and the regions above and below the bulkhead plate which divides the drywell into the upper head region and the lower region.

Two analyses were 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, determines the maximum downward loading due to pipe breaks, and the second analysis, a break of the RRC suction line, determines the maximum upward loading. Information with respect to the analyses for the upper head region and the lower region is provided below:

a. For the subcompartment analysis in the upper head region, the worst case is a double ended guillotine break in the 6" RCIC line above the RPV head at approximately elevation 595.1 ft. For the analysis in the lower region in order to determine the differential pressure across the bulkhead plate, the worst case is a double ended guillotine break in the 24" recirculation line anywhere inside the drywell. The pipe breaks were postulated for the subcompartment structural design and the component support design.

b. The blowdown mass and energy release rates as functions of time for the 6" RCIC line break are shown in Table 6.2-20 (Steam) and 6.2-21 (Water). The blowdown mass and energy release rates as functions of time for the 24" recirculation line break are shown in Table 6.2-22 (Steam) and 6.2-23 (Water).

c. The subcompartment analyses for the case of a 6" RCIC line break in the upper head region and the case of a 24" recirculation line break were performed with the computer code RELAP4/MOD5 (Reference 6.2-14). In the computer model, the drywell volume is represented by two nodes. Node 1 represents the upper head region and Node 2 represents the lower region of the drywell. For reasons of conservatism, the wet well is not modeled in the analyses. Vent paths connecting Node 1 and Node 2 pass through the bulkhead plate. For the break in the upper head region three sets of vents are considered. The three sets of vents are two open vents, two vents through backdraft dampers (setting 3IWG) and the ventilating fans, and two vents through the HVAC ducts and backdraft dampers (setting 9IWG). For the break in the lower region of the drywell two sets of vents are considered. The two sets of vents are the two open vents and two vents through backdraft dampers (setting 3IWG) and the

HVAC ducts. In the computer model backdraft damper opening delays of 1 psi and 0.25 second are assumed.

Figure 6.2-36 shows the nodalization scheme in the drywell. Figure 6.2-37(2a) shows the plan view of vents in the bulkhead plate. Figure 6.2-37(2b, 2c, 2d) show the sectional views and dimensions of the three types of vent which are being incorporated in the design.

d. The nodal volume data used for the analysis of a 6" RCIC line break in the upper head region and the analysis of a 24" recirculation line break in the lower region is shown in Table 6.2-24 Table 6.2-25 shows the flow path data for the analysis of a 6" RCIC line break and Table 6.2-26 shows the flow path data for the analysis of a 24" recirculation line break. Form loss coefficients were determined based on the data in Reference 6.2-15.

e. Since there are no significant obstructions in the proximity of the pipe break considered in the analyses, significant pressure variation in any direction is not expected. The two node model used for the analyses is considered to be adequate and a sensitively study is not necessary.

f. There are no movable obstructions in the vicinity of the vent. Insulation for piping and components was assumed to remain intact during the accident, and volume of insulation was subtracted from the nodal volumes.

g. The absolute pressure responses as a function of time in the upper head region and the lower region in the drywell are shown in Figure 6.2-38 for the case of a 6" RCIC line break, and in Figure 6.2-39 for the case of a 24" recirculation line break. Figures 6.2-40 and 6.2-41 show the pressure differential across the bulkhead plate for the case of a 6" RCIC line break, and the case of a 24" recirculation line break, respectively.

h. The peak differential pressure and the time of the peak for the case of a 6" RCIC line break and the case of a 24" recirculation line break are shown in Table 6.2-27.

Peak and transient loading in other major compartments, such as the drywell and the upper head region of primary containment were included in the basic design. Since these compartments are large and relatively unencumbered, the loads are time dependent but relatively uniform throughout the compartment in question. The time dependent loads were applied as equivalent static loads, utilizing the appropriate dynamic load factors. The component stresses were found to be within the values specified in the appropriate Codes; however, after a LOCA, the refueling bulkhead would require requalification prior to use. This is considered acceptable since the refueling bulkhead does not perform a safety-related function and would not become a missile during the postulated LOCA.

The analyses for the annulus were reported in full detail in References 6.2-9 through 6.2-11. All potential pipe breaks within the sacrificial shield wall have been evaluated. The information is contained in References 3.8-5, 3.8-6, 3.8-7, and 3.8-24. These references have been previously submitted to the NRC. The result of the case of a 60-node model of the shield wall annulus for pressure transient calculation was confirmed by the NRC, and the analysis was considered acceptable for the shield wall base design and the design of the shield wall above the base, as stated in NRC letters (References 6.2-12 and 6.2-13).

Peak and transient loading used to establish the adequacy of the sacrificial shield wall, including the time/space dependent forcing functions are presented in References 6.2-9 through 6.2-11 and 3.8-24.

Subsequently, a more realistic approach was used in determining loads from postulated pipe breaks within the annulus area. These loads were used to produce response spectra for use in evaluating the secondary effects (the dynamic effects on piping systems, equipment and components attached to the sacrificial shield wall or the RPV). Three principal changes were made in the assumptions used in the previous more conservative sacrifical shield wall analysis. Namely:

a. The volume in the annulus was utilized to receive the blowdown with the RPV insulation volume conservatively assumed not to be available.

b. A finite time dependent blowdown was used for the recirculation break, utilizing NSSS supplier's methodology. The effect of subcooling has been taken into account.

c. The feedwater pressurization analysis was developed utilizing blowdown values developed by detailed computer analysis rather than the previous hand calculation method.

Current state-of-the-art industry methods were used for these annulus pressurization calculations. These methods result in more realistic prediction of pressures as compared to the more conservative calculations discussed previously. Each of the three changes employed are described below:

a. Annular Volume

The current industry approach is to utilize the annular volume excluding the RPV insulation volume which is conservatively assumed not to be available. This approach is conservative but more realistic than previous analyses where only the annular volume on one side of the RPV insulation was available.

b. Finite Time Dependent Blowdown

The blowdown loading values given in Reference 6.2-11 were derived with the assumption that the pipe break would occur instantaneously and that the annulus area would see the maximum blowdown instantaneously ~~at the same time.~~ ~~Realistically, it is assumed that~~ Actually, the full flow from the severed pipe can not be realized until the severed pipe ends separate a distance equal to one half (1/2) the pipe diameter. Movement actually occurs in a finite time and is a function of the stiffness characteristics of the pipe and the restraining capability of the pipe whip restraints.

Current industry practice was used to develop displacement versus time data for a finite break opening; the General Electric analytical method for determining the short-term mass and energy release was used. The analysis was utilized for the recirculation loop break, but not for the feedwater line since it was determined that the small percentage reduction for the feedwater would not warrant the additional calculations.

c. Feedwater Break Blowdown Data

The blowdown analysis for the postulated feedwater line break was based on a comprehensive model developed for the entire ~~condensate~~ feedwater system from the condenser to the reactor vessel. This model, in conjunction with the RELAP 4/MOD 5 computer program (Reference 6.2-14) was used to calculate the transient and energy blowdown data.

CONTAINMENT SYSTEMS BRANCH

ISSUE 37

NRC: Subcompartment Pressurization. The information provided to date is not sufficient to allow the NRC to perform a confirmatory analysis. See also Question 022.005.

Supply System: The Supply System will verify the response to Question 022.005 for adequacy, and check Susquehanna FSAR Appendix 6A for applicability to WNP-2. Either Burns & Roe or General Electric will provide force vs. time calculation results in the reactor vessel for overturning. The Supply System has provided the copies of the Subcompartment Pressurization reports. The NRC required additional information regarding the forces and moment and will be supplied by October 2, 1981.

Summation: F. Eltawila will perform a confirmatory analysis based on the Supply System response to Question 022.005, and the reports to be provided by the Supply System.

Supplementary was information provided on January 6, 1982, (G02-82-03, attached).

THIS LETTER SATISFIES COMMITMENT NO. ______
THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.
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G02-82-03
SS-L-02-CDT-82-001

Docekt No. 50-397

Mr. A. Schwencer, Director
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SUBCOMPARTMENT PRESSURIZATION

Attached are sixty (60) copies of the information on subcompartment pressurization requested by the NRC in the Containment Systems Branch meeting held September 14-17, 1981. This information is in response to Issue #37 from the minutes of that meeting.

Very truly yours,

Original signed by:

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca
Attachment

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

FILE COPY

AUTHOR:	CD Taylor		FOR SIGNATURE OF:	GD Bouchey	
SECTION					
FOR APPROVAL OF	RM Nelson	BA Holmberg	GC Sorensen		
APPROVED	RM Nelson	DC Timmins	for GC Sorensen		
DATE		1/5/82	GCS 1/5/82		

CONTAINMENT SYSTEMS BRANCH

ISSUE 38

NRC:	II.F.1.4 II.F.1.5 II.F.1.6	Provide accuracy of instrumentation and response time.
	II.F.1.4	Provide -5 psig indication capability.
Supply System:	The information will be given in conjunction with Regulatory Guide 1.97 response in early January 1982.	
	A response to this issue was provided in the TMI (Appendix B) submittal.	

CONTAINMENT SYSTEMS BRANCH

ISSUE 39

NRC: When will we submit an amendment to the DAR? NRC has received Revision 1 or 2 dated September 1979.

Supply System: The load definitions section information was submitted as a DAR appendix January 13, 1982 in letter G02-82-34.

The DAR will be revised and resubmitted prior to fuel load to close open SER issue 3.8.2(d).

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

January 13, 1982
G02-82-34
SS-L-02-CDT-82-014

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
APPENDICES TO WNP-2
DESIGN ASSESSMENT REPORT (DAR)

Enclosed are sixty (60) copies of Appendices H and I to the Design Assessment Report for WNP-2. These appendices are being submitted to NRC in draft form at this time and will be incorporated into a DAR amendment prior to July 1, 1982.

Very truly yours,

G. D. Bouchey, Deputy Director
Safety & Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Reil - NRC Site

CONTAINMENT SYSTEMS BRANCH

ISSUE 40

NRC:	Add a column to Table C.1 of NUREG-0808 indicating compliance. Also expand Table C.1 to include SRV loads from NUREG-0487 and Supplement 1 to NUREG-0487.
Supply System:	Table C.1 of NUREG-0808 has been updated to show WNP-2 compliance with generic position acceptable to NRC on plant unique application. For downcomer lateral loads, NRC was notified in Table C.1 of potential need for review to permit consideration of the probability of peak loads occurring simultaneously per 2.3.3.2.3 in draft NUREG-0808. Also, a Table from NUREG-0487 and supplement 1 to NUREG-0487 has been attached and marked to indicate the position for WNP-2.

TABLE C-1 MARK II LOCA-RELATED HYDRODYNAMIC LOADS

SUMMARY TABLE

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II O.G. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
A. Submerged Boundary Loads During Vent Clearing	24 psi overpressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface.	Letter No. 00-79 from GE to NRC dated 3/20/79	----	(2)*II.A.1	Acceptable
B. Pool-Swell Loads					
1. Pool-Swell Analytical Model					
a) Air-Bubble Pressure	Calculated by the pool-swell analytical model (PSAM) used in calculation of submerged boundary loads.	NEDE/NEDO-21061 NEDE-21544-P	----	(1)III.B.3.a.1	Acceptable
b) Pool-Swell Elevation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the elevation corresponding to the drywell floor uplift ΔP=2.5 psid.	Letter No. 360 from LILCO to NRC dated 2/16/79	A.1	(2)II.A.2	Acceptable

* (1) - Reference NUREG-0487
(2) - Reference NUREG-0487 Supplement 1
(3) - Reference NUREG-0808

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II O.G. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
c) Pool-Swell Velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady-state drag between vent exit and maximum pool elevation. Analytical velocity variation is used up to maximum velocity. Maximum velocity applies thereafter up to maximum pool swell. PSAM predicted velocities multiplied by a factor of 1.1.	NEDE/NEDO-21061 NEDE-21544-P	A.2	(1)III.B.3.a.3	Acceptable
d) Pool-Swell Acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration loads on submerged components during pool swell.	NEDE/NEDO-21061 NEDE-21544-P	----	(1)III.B.3.a.4	Acceptable
e) Wetwell Air Compression	Wetwell air compression is calculated by PSAM consistent with maximum pool swell elevation in B.1.b above.	Letter No. 360 from LILCO to NRC dated 2/16/79	----	(2)II.A.2	Acceptable

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II D.O. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
f) Drywell Pressure	Methods of NEDM-10320 and NEDO-20533 Appendix B. Utilized in PSAR to calculate pool swell loads.	Letter No. 002 from LILCO to NRC dated 6/11/81 NEDM-10320 NEDO-20533 App. B	----	(1)III.B.3.a.6	Acceptable
2. Loads on Submerged Boundaries	Maximum bubble pressure predicted by the PSAR added uniformly to local hydrostatic below vent exit (walls and basemat) linear attenuation to pool surface. Applied to walls up to maximum pool swell elevation.	NEDE/NEDO-21061 NEDE-21544-P	----	(1)III.B.3.b	Acceptable
3. Impact Loads					
a) Small Structures	1.35 x Pressure-Velocity correlation for pipes and I beams based on PSTF impulse data and flat pool assumption. Variable pulse duration.	----	A.5	(1)III.B.3.c.1	Acceptable
b) Large Structures	None - Plant unique load where applicable.	----	----	(1)III.B.3.c.6	Acceptable (No large structures for WNP-2)
c) Grating	P drag vs. grating area correlation and pool velocity vs. elevation. Pool velocity from the PSAR. P drag multiplied by dynamic load factor.	NEDE/NEDO-21061	A.3	(1)III.B.3.c.3	Acceptable

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II O.G. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
4. Wetwell Air Compression					
a) Wall Loads	Direct application of the PSAR calculated pressure due to wetwell compression.	NEDE/NEDO-21061	----	(1)III.B.3.d.1	Acceptable.
b) Diaphragm Upward Loads	5.5 psid for diaphragm loadings only.	Letter No. MFN-125-81 dated 6/30/81	----	(3)2.1.2.7	Acceptable
5. Asymmetric LOCA Pool	Use 20 percent of maximum bubble pressure statistically applied to 1/2 of the submerged boundary.	Letter No. 76-79 from GE to NRC dated 3/16/79	A.4	(2)II.A.3	Acceptable
C. Steam Condensation and Chugging Loads					
1. Downcomer Lateral Loads					Acceptable If necessary WNP-2 may seek load reduction by considering probability of peak loads occurring simultaneously as provided in draft NUREG-0808
a) Single-Vent Loads (24 in.)	Dynamic load to end of vent. Half sine wave with a duration of 3 to 6 ms and corresponding maximum amplitudes of 65 to 30 Klbf.	NEDE-23806-P	B.1.a	(3)2.3.3.2	
b) Multiple-Vent Loads (24 in.)	Prescribed variation of load per vent vs. number of vents. Determined from single vent dynamic load specification and multivent reduction factor.	Letter No. 077-80 from GE to NRC dated 4/9/80	----	(3)2.3.3.3	
c) Single/Multiple vent loads (28 in.)	Multiply basic vent loads by factor f=1.34	Letter No. MFN-012-81 from GE to NRC dated 1/16/81	B.1.b	(3) 2.3.2.1	

WNP-2 position is that CO does not represent a governing load, ~~for WNP-2~~ and need not be considered in assessments of structures, piping, and equipment, as discussed in letter G02-81-552 and Report attached thereto.

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II D.O. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
2. Submerged Boundary Loads					
a) High/Medium Steam Flux Condensation Oscillation Load	Bounding CO pressure histories observed in 4TCO tests. Inphase application.	NEDE-24288-P	----	(3)2.2.1.3	~~WNP2 position is that chugging bounds CO and was transmitted to the NRC via letter G02-81-239 dated 8/13/81, with some supporting evidence. Consequently, a plant designed for chugging load effects need not be assessed for CO load effects. A report substantiating this statement will be submitted before Jan. 1982~~
b) Low Steam Flux Chugging Load	Conservative set of 10 sources derived from 4TCO tests. Applied to plants using the IWEGS/MARS acoustic model. Source desynchronization of 50 ms or alternate load using sources derived from the 4TCO key chugs without averaging.	NEDE-24302-P NEDE-24022-P Letter No. MKII-085-LIC from LILCO to NRC dated 7/14/81	----	(3)2.2.2.3	Plant unique. WNP-2 chugging report submitted July 1981
- Symmetric Load	All vents utilize source of equal strength for each of the sources.				
- Asymmetric Load Case	Source strengths Si = S (1±α) applied to all vents on + and - side of containment. Sources based on the symmetric sources. Asymmetric parameter α based on rms moment method of interpreting experimental 4TCO single-vent and JAERI multivent data.				

LOAD OR PHENOMENON	LOAD SPECIFICATION	MARK II D.B. DETAILED LOAD DESCRIPTION	NRC ACCEPTANCE CRITERIA	NRC EVALUATION	WNP-2 POSITION
D. Secondary Loads					
1. Sonic Wave Load	Negligible load	NEDE/NEDO-21061	----	(1)III.E.1	Acceptable
2. Compressive Wave Load	Negligible load	NEDE/NEDO-21061	----	(1)III.E.2	Acceptable
3. Fallback Load on Submerged Boundary	Negligible load	NEDE/NEDO-21061	----	(1)III.E.5	Acceptable
4. Thrust Loads	Momentum balance	NEDE/NEDO-21061	----	(1)III.E.6	Acceptable
5. Friction Drag Loads on Vents	Standard friction drag calculations	NEDE/NEDO-21061	----	(1)III.E.7	Acceptable
6. Vent Clearing Loads	Negligible load	NEDE/NEDO-21061	----	(1)III.E.8	Acceptable
E. LOCA Submerged Structure Drag Loads During The Water Jet and Air Bubble Periods	Calculated by the Ring Vortex Model	Ring Vortex Model Report dated September 1980		(3)2.2.4.3	Acceptable with below clarification agreed upon with NRC during meeting of Sept. 16, 1981

Both the Ring Vortex Model and the LOCA bubble charging model will be used and the largest (maximum) calculated flow field (velocity and acceleration) values anywhere in the pool will be used to define loads on submerged structures.

Table IV-1 Alternative Mark II Lead Plant Pool Dynamic Loads

Load or Phenomenon	Mark II Owners Group Alternative Load Specification	Reference	NRC Review Status	Section in this Report	WNP-2 POSITION
I. LOCA Related Hydrodynamic Loads A. Submerged Boundary Loads During Vent Clearing	24 psi overpressure statically applied with hydrostatic pressure to surfaces below vent exit (attenuate to 0 psi at pool surface) for period of vent clearing.	March 20, 1979 GE letter (3)	Acceptable	II.A.1	See Table C-1 of NUREG-0808
B. Pool Swell Loads 1. Pool Swell Analytical Model (PSAM) a) Wetwell Air Compression b) Pool Swell Elevation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the elevation corresponding to the drywell floor uplift ΔP per NUREG 0487 criteria I.A.4. The associated maximum wetwell air compression is used for design assessment.	February 16, 1979 Shoreham letter (6)	Acceptable	II.A.2	
2. Asymmetric LOCA Pool Boundary Loads	Use 10% of maximum bubble pressure statically applied to 1/2 of the submerged boundary.	March 16, 1979 GE letter (9)	Use 20% of maximum bubble pressure statically applied to 1/2 of the submerged boundary.	II.A.3	
II. SRV Related Hydrodynamic Loads					
A. Methodology for T-Quencher Load Prediction	Interim T-Quencher Load Definition	Susquehanna DAR (13)	Acceptable with the following modifications • Bubble Frequency - 3 to 11 Hz • Peak Pressure Multiplier for Subsequent Actuation - 1.5 • Vertical Pressure Profile - maximum amplitude from basemat to 2.5 ft above quencher center line, linear attenuation to zero at pool surface. • Multiple SRV Actuations - 1) linear ABSS superposition of peak single valve values with all bubbles in phase	II.B.5	Not applicable to WNP-2 (which utilizes X-Quencher)
B. Methodology for X-Quencher Load Prediction	WNP-2 plant unique improved X-quencher load definition based on CAORSO test data	SRV loads report dated 7/29/80, submitted to NRC in August 1980	Acceptable with comments as discussed in WPPSS/NRC meeting of Sept. 15&16 1981	NA	Acceptable

Table IV-1 Alternative Mark II Load Plant Pool Dynamic Loads (Continued)

Load or Phenomenon	Mark II Owners Group Alternative Load Specification	Reference	NRC Review Status	Section in this report	WNP-2 POSITION
			2) If the combined peak pressure exceed local single valve peak use the lower value		
III. LOCA/SRV Submerged Structure Loads					
A. Air Bubble Loads					
1. Standard Drag in Accelerating Flow Fields	Draft Coefficients are presented in Attachment 1.k of the Zimmer FSAR	Attachment 1.k Zimmer FSAR [20]	Acceptable with the following modification 1) Use $C_M = C_M - 1$ in the F_A formula. 2) For non cylindrical structures use lift coefficient for appropriate shape or $C_L = 1.6$ 3) The standard drag coefficient for pool swell and SRV oscillating bubbles should be based on data for structures with sharp edges.	II.C.2	Generic methodology acceptable. Plant unique flow fields are consistent with C.2.b from Table C-1 of NUREG-0808 for LOCA loads and with II. from Table IV-1 of NUREG-0487 Supplement 1 for SRV loads. (Amplitudes for SRV loads verified by CAORSO data on submerged structures)
2. Equivalent Uniform Flow Velocity and Acceleration	Structures are segmented into small sections such that 1.0 < L/D < 1.6. The loads are then applied to the geometric center of each segment.	Attachment 1.k Zimmer FSAR	Acceptable	II.C.2	
3. Interference Effects	A detailed methodology is presented in Attachment 1.k of the Zimmer FSAR	Attachment 1.k Zimmer FSAR	Acceptable	II.C.2	

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	SER Section	WNP-POSIT
II. SRV-Related Hydrodynamic Loads					
A. Pool Temperature Limits for KWU and GE four arm quencher	None specified	N/A	NRC Criteria II.1 and II.3	III.C.1	
B. Quencher Air Clearing Loads	Mark II plants utilizing the KWU quencher use an interim load specification consisting of the ramshead calculational procedure. Mark II plants utilizing the four arm quencher use quencher load methodology described in DFFR.	DFFR Revision 2	NRC Criteria II.2	III.C.2.b III.C.2.c	
C. Quencher Tie-Down Loads					
1. Quencher Arm Loads					
(a) Four Arm Quencher	Vertical and lateral arm loads developed on the basis of bounding assumptions for air/water discharge from the quencher and conservative combinations of maximum/minimum bubble pressure acting on the quencher.	DFFR Revision 2	Acceptable	III.C.2.e.1	Accep
(b) KWU T Quencher	KWU "T" quencher not included in Mark II O.G. Program. T quencher arm loads not specified at this time.	N/A	Review Continuing		N.A

N

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	SER Section	WNP-2 POSITIO
2. Quencher Tie-Down Loads					
(a) Four-Arm Quencher	Includes vertical and lateral arm load transmitted to the basemat via the tie downs. See II.C.1.a above plus vertical transient wave and thrust loads. Thrust load calculated using a standard momentum balance. Vertical and lateral moments for air or water clearing are calculated based on conservative clearing assumptions.	DFFR Revision 2	Acceptable	III.C.2.e.2	Accepta
(b) KWU "T" Quencher	KWU "T" quencher not included in Mark II O.G. program. T quencher tie-down loads not specified at this time	N/A	Review Continuing		N.A.

IV-7

Q. 130.050
(Q220.001)
(3.5.1)

You state in Section 3.5.1.3 of the FSAR that the reorientation of the turbine generator building to limit potential missile strike is not considered. Rather, the barrier capability of the massive radiation shielding structures, characteristic of BWRs, is utilized to control postulated turbine missile hazards, and probability studies provide the assurance that the chance of missile strike is remote.

Describe your probability studies with emphasis on the chance of turbine missile strike and penetration of the structural barrier. If in your analysis the value of P3 is assumed as 1.0, please so indicate.

Response:

WNP-2 has completed a turbine missile study consisting of a probabilistic approach to missile strikes and damage.*

*Revised FSAR page changes attached.

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

January 13, 1982
G02-82-33
SS-L-02-CDT-82-013

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
NRC QUESTION 130.050
TURBINE MISSILE STUDY

Enclosed are sixty (60) copies of the draft response to NRC Question 130.050 and revised WNP-2 FSAR pages. This response shows the results of the turbine missile study for WNP-2.

All enclosed information will be incorporated into the WNP-2 FSAR in Amendment 23.

Very truly yours,

G. D. Bouchey, Deputy Director
Safety & Security

CDT/jca
Enclosures

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

Open SER Issue

ICSB-Item 2

Concern:

Address R.G. 1.97 on an item-by-item basis by submitting a revised FSAR Section 7.5 by 12.31.81.

Response:

A response to this issue was submitted on January 13, 1982 in letter G02-82-30.

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

January 13, 1982
G02-82-30
SS-L-02-CDT-82-010

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
CHAPTER 7.5 REWRITE

Enclosed are sixty (60) copies of the WNP-2 revised FSAR Section 7.5, which addresses Regulatory Guide 1.97 on an item-by-item basis. This issue was identified as ICSB-2 at the branch meeting September 25, 1981 and closes the open SER issue.

These revised FSAR pages will be included in Amendment 23 to the WNP-2 FSAR.

Very truly yours,

G. D. Bouchey, Deputy Director
Safety & Security

CDT/ct
Enclosure

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

ICSB-Item 3

Concern:

WNP-2 was requested to provide a functional description of the multiplexers used to relay system status (parameters) to the control room and to provide reliability information.

Response:

The attached article provides the functional description and reliability information about the multiplexers.

REMOTE MULTIPLEXING SYSTEM APPLICATION TO A NUCLEAR GENERATING STATION - AN UPDATE

ICSB/No. 3 (A1 320 - Dwg - No. 5527)

O.S. Mazzoni, P.E., Senior Member, IEEE
A.L. Cava, P.E., Member, IEEE
G.D. Bijoor, P.E., Member, IEEE

Burns and Roe, Inc.
Woodbury, New York

M.F. Wiitala, P.E., Member, IEEE

Washington Public Power Supply System
Richland, Washington

Abstract - A previous paper (1) gave a preliminary description of a Remote Multiplexing System (RMS) for the Washington Public Power Supply System's Nuclear Project No. 2 in its initial design stage. This paper describes the final design including Class IE qualification requirements and testing. It is believed that this is the first time a RMS design has been used in Class IE systems and for such extensive control and data gathering applications to achieve substantial economy and flexability in a nuclear power generating station. It is hoped that valuable experience will be obtained from this installation and will pave the way for further RMS applications.

INTRODUCTION

Washington Public Power Supply System's Nuclear Power Plant WNP-2 is a 1200 MW, boiling water reactor (BWR) nuclear generating facility now under construction at Hanford, Washington. This plant has been designed by Burns and Roe, Inc. and is scheduled to begin commercial operation in 1980.

An early study indicated that substantial savings in cost could be obtained by the substitution of the Remote Multiplexing System (RMS) for the control and signal wiring between the Control Building and the outside facilities, which required long cable runs. This feature and the advantage of the flexibility for future expansion of the system led to the acceptance of the RMS for this nuclear generating facility.

The purpose of this paper is to give the details of the RMS used in the WNP-2 project. It discusses the qualification requirements for Class IE systems and includes the description of the tests on the system. The paper includes an Appendix A which gives the most commonly used terms in a basic multiplexing system.

GENERAL DESCRIPTION

The power plant is located three miles west of the Columbia River in the Hanford Reservation in the State of Washington. It consists of the Reactor Building surround-

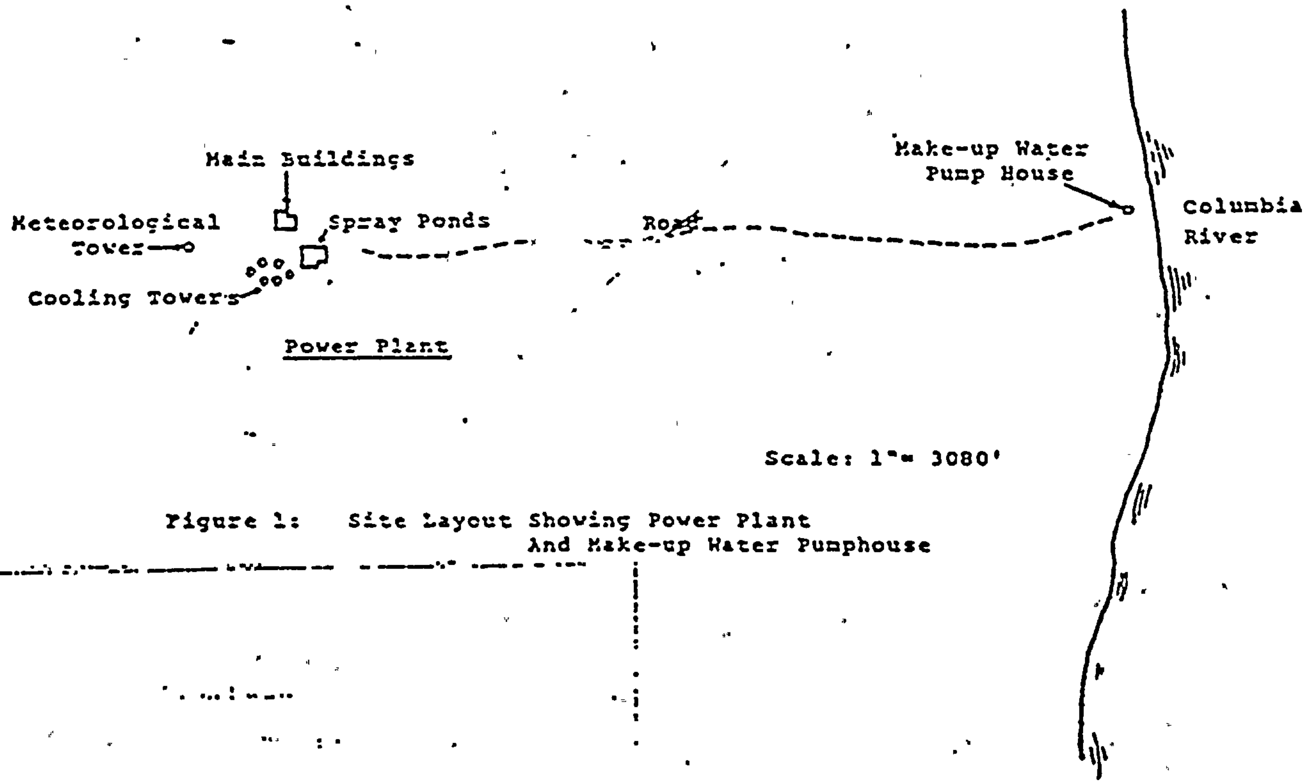

Figure 1: Site Layout Showing Power Plant And Make-up Water Pumphouse

No. 3 1/9

ed by structurally independent buildings. Individual buildings house the Turbine-Generator, Radwaste Handling Facility, Emergency Diesel Generators, Control Room and Service Building. All the controls and instrumentation from these buildings are connected to the Control Building by hardwired connections.

The outside facilities consisting of the Circulating Water Pumphouse, two Standby Water Pumphouses, six Cooling Towers, Make-up Water Pumphouse and the Meteorological Tower are connected to the Control Building by the RMS. Figures 1 and 2 show the arrangement of the structures.

The Circulating Water Pumphouse contains three 5060 HP circulating water pumps, two 1500 HP plant service water pumps, two electric fire pumps, one diesel driven fire pump, a jockey pump, a water treatment system and numerous valves. These are controlled and monitored from the control room via the RMS.

For cooling the circulating water flow, there are six circular mechanical draft cooling towers with (36) 200 HP fans. Ice control is accomplished by reversing motors. Inlet and outlet water temperatures of the cooling towers are monitored by thermocouples.

For emergency cooling, 13 million gallons of water are stored in two 1½ acre spray ponds. Adjacent to each spray pond is a pumphouse with standby pumps to provide emergency cooling service water to the reactor and the three emergency diesel generators. These pumps are Class IE.

Located about three miles east of the nuclear power plant at the Columbia River is the

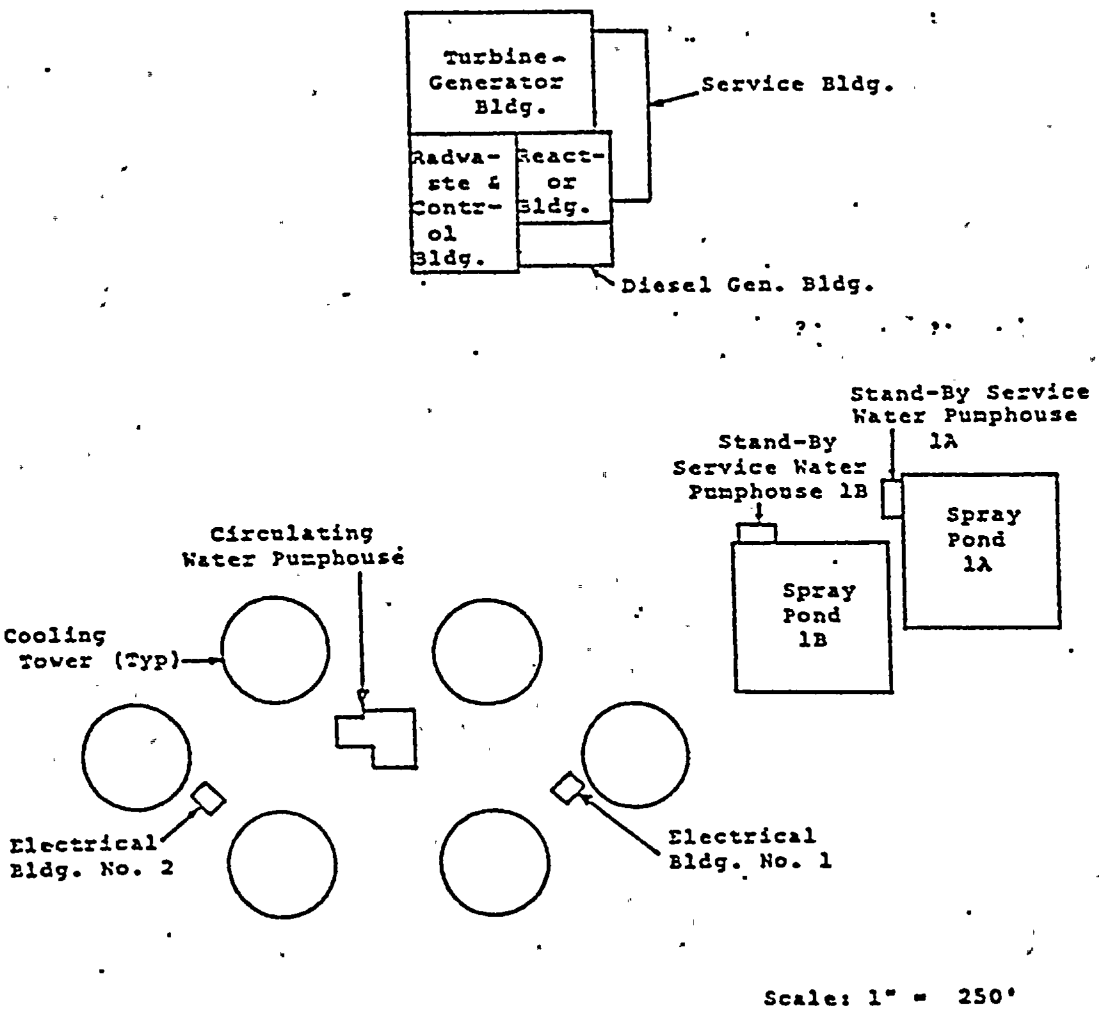

Figure 2: Arrangement of Main Power Plant Structures

No. 3 2/9

Make-up Water Pumphouse where three 800 HP pumps provide 25,000 GPM of make-up water to the circulating water and spray pond systems. The make-up water is required to replace water lost by evaporation.

The meteorological tower collects weather data for this plant. In the future it will also serve two other nuclear generating facilities located in the vicinity. The RMS will be extended to serve these additional plants without the need for installing additional transmitters.

Among the parameters that are monitored and/or controlled as applicable are pond and river levels, water temperatures, pressures, thermocouples, valve positions and motor conditions.

The distances from these components to the control room vary from eleven hundred feet to three miles.

QUALIFICATION REQUIREMENTS OF THE SYSTEM

The class IE RMS equipment was specified to conform to IEEE Std. 323 (2) which is a general guide for qualifying Class IE electric equipment for Nuclear Power Generating Stations. This standard includes specification requirements for type tests to simulate the service conditions, analysis and documentation. The Class IE RMS was specified to conform to IEEE 344 (3) which describes the Seismic Qualification requirements. This document provides direction for establishing procedures that will yield data to verify that the Class IE equipment can meet its performance requirements during and following one SSE (Safe Shutdown Earthquake) preceded by a number of OBES (Operating Basis Earthquakes).

Full conformance was obtained with the U.S. Nuclear Regulatory Commission (NRC) Code of Federal Regulations 10 CFR 50 (4), Appendix B. This code describes the quality assurance criteria for the equipment used in Nuclear Power Plants and details the fabrication, documentation and quality control procedures for the same.

DETAILED DESCRIPTION OF THE REMOTE MULTIPLEXING SYSTEM

General

The Remote Multiplexing System was developed and manufactured by Anaconda-I/C Engineering of Los Angeles. The system is known as Uniplex Data System, Model 600. It is a high speed, data system providing time-sharing digital communication between the main control room and the remote buildings. The RMS is designed to provide redundant paths for all critical signals.

Figure 3 shows the basic block diagram for the Multiplex Data System. Each control board instrument or device is hardwired to a terminal strip in the Control Room Multiplexer (CRM). The CRM is commanded by a Central Control Unit (CCU), point-by-point, to transmit or receive data from the associated Remote Multiplexing Panel (RMP). In the field, the process sensors or control contacts are hardwired to the termination assemblies of the RMP. The field data on temperatures, pressures, flow rates, valve positions, etc. are digitized in the RMP; stored and transmitted to the CCU once every scan cycle. The data then are fed to the CRM's which reconvert the data into their original field equivalent for use in control board instrumentation.

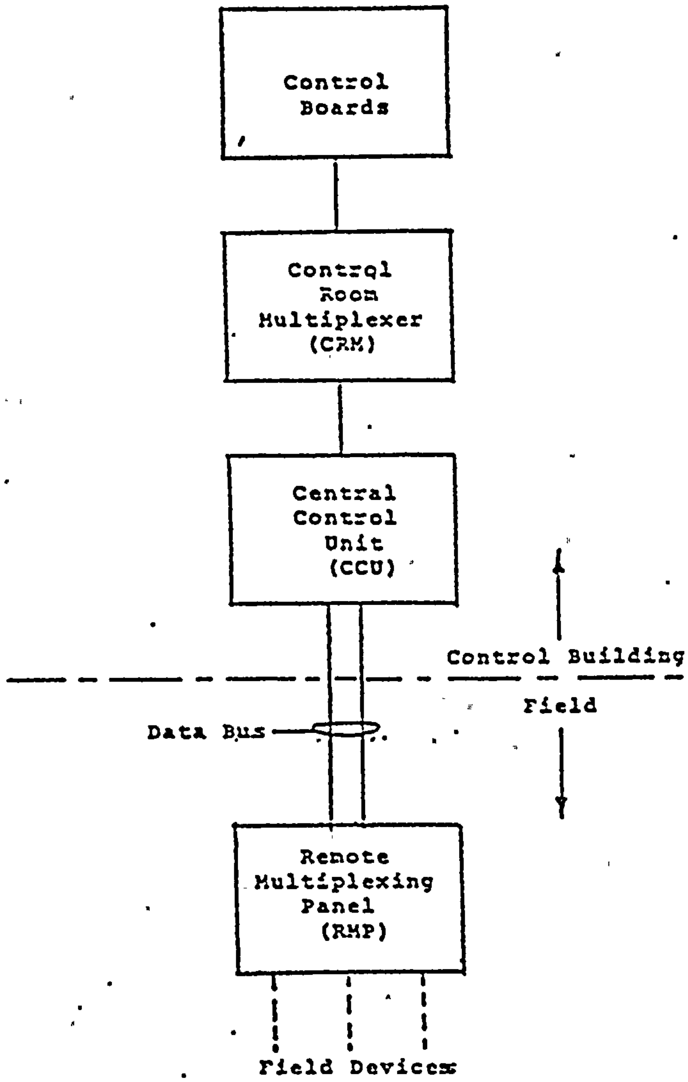

Figure 3: Basic Block Diagram For The "Uniplex Data System"

Details of the System

The RMS as offered can have up to 64 RMP's and the same number of CRM's. Each RMP and CRM includes a power supply for the electronic circuitry. For the system supplied, each RMP can accommodate a maximum of 17 function cards and a CRM has a maximum capacity of 14 function cards.

The RMS is capable of handling analog signals, contact status, and contact control signals through the use of the appropriate function cards in both the control room and field multiplexing panels.

No. 3 3/9

Figure 4 shows the details of the transmission of the analog signals to the computer, recorders, or temperature indicators. The input may be from thermocouples, RTD's, or process transducers (4 to 20mA). Temperature compensation for the thermocouples is provided at the RMP's so that it is not necessary to provide the same at the control room computer or recorders.

Analog control signals between the ranges of 4 to 20mA can also be transmitted from the control room to the remote devices for the operation of the devices such as electronpneumatic converters. Table 1 provides data on the function cards. The equipment has the capability of handling up to 300 analog signals. Space is available for additional cards if required in the future.

Contact status signals can be transmitted from the field to the control room for indicating lights, circuit interlocks, etc. This is indicated in Figure 5, which also shows the transmission of the contact control signals form the control room to the field for the actuation of the control devices such as valves, circuit breakers, etc. The RMS can handle 900 contact status/contact control signals.

The equipment can operate between 0° and 70°C temperature and humidity of 5 to 90 percent without condensation. It can withstand 10 mr/hr of radiation during operation.

Each signal is independently isolated and provides between 90 and 100dB of common mode rejection. at a signal level of 400V AC, 60 Hz. The field equipment can withstand indefinite imposition of ± 1500V of common mode voltage, whereas the control room devices have a ± 600 Volt withstand capacity.

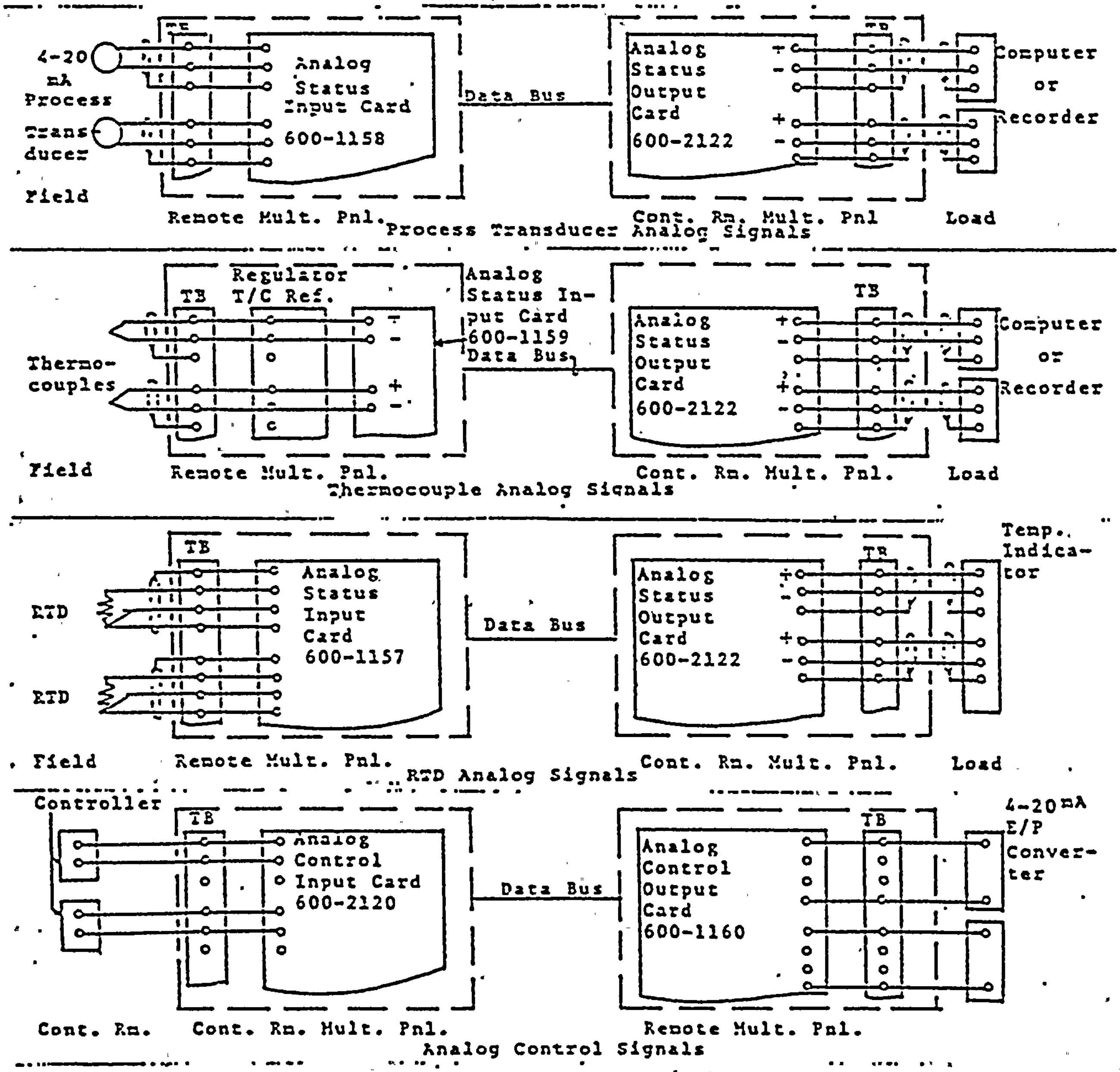

Figure 4: Transmission of Analog Signals

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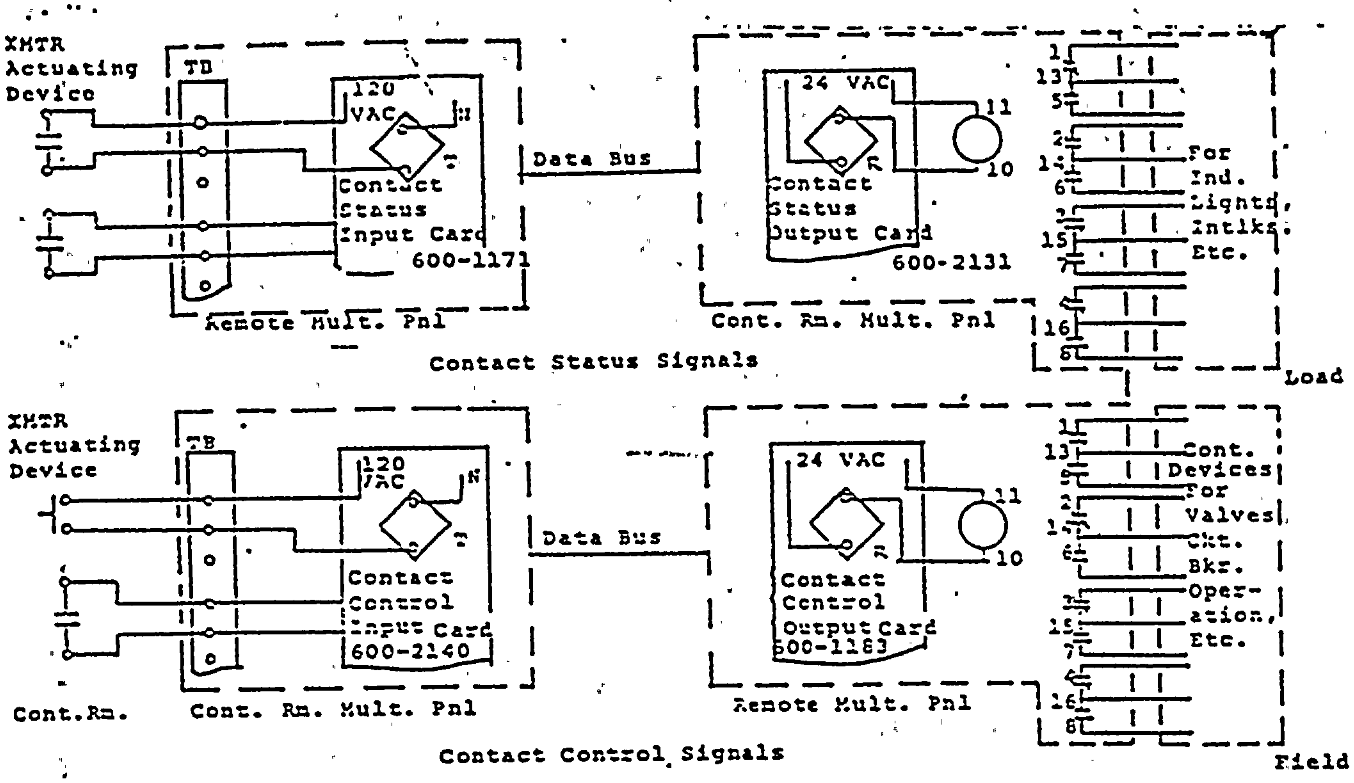

FIGURE 5: Transmission of contact status/contact control signals.

TABLE I: DATA ON FUNCTION CARDS

INPUT CARDS

CARD TYPE	RANGE	NO. OF SIGNALS PER CARD
CURRENT INPUT STATUS	4-20 mA	8
VOLTAGE INPUT STATUS	0-5 V	8
T/C INPUT STATUS	0-5 mV	8
RTD INPUT STATUS	100-136.6 Ω	8
CURRENT INPUT CONTROL	4-20 mA	6
CONTACT STATUS INPUT	-	12
CONTACT CONTROL INPUT	-	12

OUTPUT CARDS

CARD TYPE	RANGE	NO. OF SIGNALS PER CARD
CURRENT OUTPUT STATUS	4-20 mA	4
VOLTAGE OUTPUT STATUS	0-5 V	4
T/C OUTPUT STATUS	0-5 mV	4
RTD OUTPUT STATUS	4-20 mA	4
CURRENT OUTPUT CONTROL	4-20 mA	4
CONTACT STATUS OUTPUT	-	12
CONTACT CONTROL OUTPUT	-	12

Method of Operation

Two sets of messages are assembled and decoded by the Remote Multiplexing Panels and the Central Control Units. Transmissions to a RMP are called interrogations and are used for requesting data and for sending commands. Transmissions from a RMP are called responses and are used for obtaining data and command confirmations where applicable.

Interrogation messages originate in the Control Room Multiplexers (CRM). The CCU modulates them into bifrequency FSK (frequency shift keying) for transmission to the selected RMP. The returned response message is transmitted using biphase PSK (phase shift keying); is demodulated and passed on to the CRM. The use of these two modulation methods prevents a response message from being interpreted as an interrogation during outside electrical interference.

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The CCU polls a given RMP by address, designates the point to be selected, prescribes the operation (Analog/Digital conversion, or status monitor), and then waits a fixed interval to receive a confirming response. Upon receipt of the response, the responding address is compared to the interrogation; and the data are accepted.

The interrogation/response message has 40 bits. Each interrogation message includes 16 bits for synchronization error detection, 8 or 12 bits of the output command data, and 12 or 16 bits for address decoding.

Class IE Equipment

Figure 6 is a block diagram showing the RMS used for the Standby Service Water Pumphouses. It is divided into three sections. This is specified, designed and fabricated for Class IE nuclear classification.

The data bus is made up of two twin-axial cables and the data are transmitted at the rate of 924,000 bits/second. The special twin-axial cable is a design requirement for the high speed data transmission. The data bus cables are run entirely in conduits.

The Division I system has its RMP located in the Pumphouse 1A and a CCU located in the control room. The scan cycle time for Division I is 120 milliseconds.

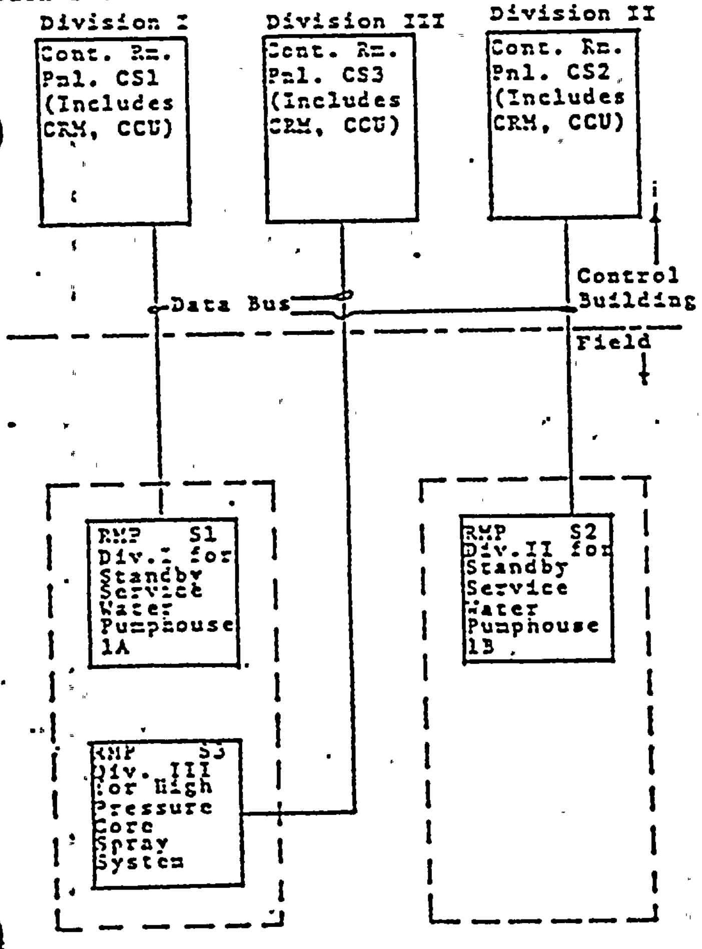

Figure 6: Block Diagram of RMS For Class IE Equipment

Standby Service Water Pumphouse 1B is served by the Division II system which is redundant to Division I. The CCU is located in the remote shutdown room in the control building and the RMP is located in the Standby Service Water Pumphouse 1B. The scan cycle time for Division II is also 120 milliseconds.

The Division III system provides controls for the high pressure core spray system. The RMP is located in the Pumphouse 1A and the CCU is located in the control room. It has a scan cycle time of 40 milliseconds.

Non-Class IE Equipment

The block diagram showing the non Class IE equipment is given by Figure 7. The system comprises of four remote multiplexing panels in Division A and five remote multiplexing panels in Division B. In this case, a single CCU in each system provides control of communications between the control boards and the remote multiplexing panels. The control room multiplexing panels and the remotes are connected by two (2) pairs of cables. The cable used for direct burial has neoprene jacketing and rodent proof shielding. An impedance matching switching device is provided for each division to address signals to and from a Receiver/Transmitter unit to be directed to a particular data bus. The rate of data transmission is 9600 bits/second. The slower speed as compared to that for Class IE system has resulted in cost savings for the equipment. The scan cycle time for each of Division A and Division B is less than 2.5 seconds.

RELIABILITY OF THE SYSTEM

A number of design features add to the reliability of the system. The major ones are summarized below. Data on reliability analysis are also provided.

Automatic Repeat Request (ARQ) Feature

The CCU polls a given RMP, designates the point to be selected and prescribes the operation to be performed (A/D conversion, or status monitor). After a fixed interval, a response is received. The response is compared to the interrogation and the data are accepted. Failure to receive a response after the fixed time interval or lack of confirmation in any portion of the response address indicates a problem either in the data system or in the RMP. This results in a repeat of the interrogation up to two times. If response confirmation is still not received, a "no response" alarm is generated in the control room. Steps can then be taken to diagnose and analyze the problem.

Error Rates and Noise Rejection

Data supplied by the manufacturer give an error rate of once every 67,000 [5] years for the Class IE high speed systems and 1.3 x 10^7 years for the non-Class IE low speed systems.

The method used in the transmission of the messages provide added protection against electrical interference by preventing a response message from being interpreted as an interrogation.

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were used for other analog tests.

Thermocouple signals were tested by providing millivolt inputs to the thermocouple cards by means of a digital potentiometer. As before, input and output values were measured by digital voltmeters.

Testing for RTD signals was performed using a resistor box for providing the input. The outputs were measured by digital voltmeters.

Testing of the contact status and contact control signals was done by shorting the input terminals by a jumper and verifying its corresponding contact closure with an ohmmeter. The tests confirmed correct operation of the output devices.

There were isolated cases of non-functional relays, faulty socket wiring and one case of an analog functional input card which needed an adjustment in its circuit. The relays were replaced, the socket wiring was corrected and the circuit of the input card was adjusted.

Tests were also performed to verify the diagnostic and alarm annunciation capabilities of the equipment. These included power failure tests, stuck bit and scan time diagnostic tests and panel alarm tests.

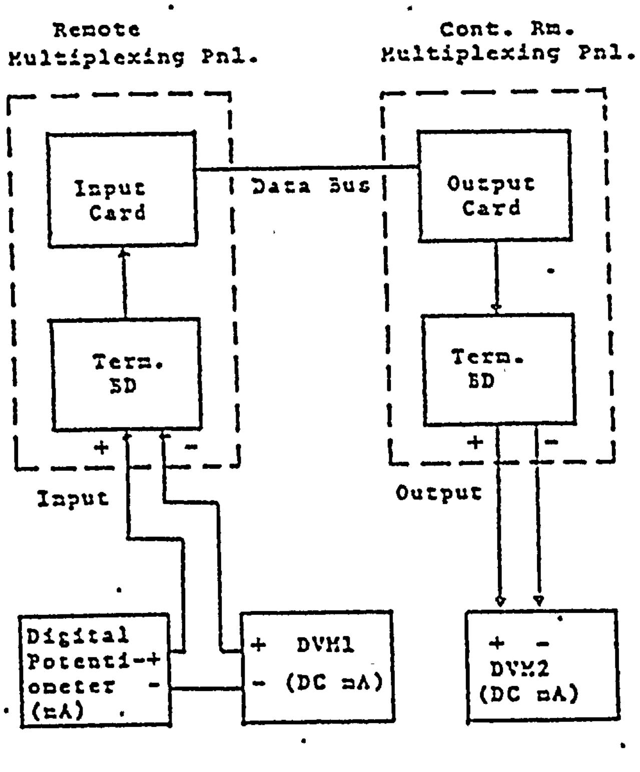

Figure 8: Testing of 0-20 mA Analog Signals

Seismic Testing

The equipment was seismically type-tested to the requirements of the Seismic Category I.

Seismic Class I equipment is designed such that there is no loss of safety-related equipment function during and after the prescribed seismic disturbance. The tests were performed in accordance with the requirements of IEEE 344.

The RMS panels were bolted, through their normal mounting holes, to a one-inch thick steel plate which was welded to the test table.

Functional testing was done during the seismic testing. Monitoring of sixteen channels of relay contact "chatter" and recording four channels of analog data was done during seismic tests.

After testing in the first biaxial plane, the specimens were rotated 90° about their vertical axes for testing in the second biaxial plane.

Two control accelerometers, one for each direction of motion, were centrally mounted on the test table to provide control points. The test response spectra (TRS) were plotted from the data produced from these control accelerometers. Three response accelerometers were mounted on each cabinet, two at the top in line with the directions of motion and one on the relay mounting panel perpendicular to the panel plane. The acceleration data were formatted as spectra plots. The specimens were subjected to biaxial random motions in the two separate principal perpendicular biaxial planes.

These Spectra plots were compared to the spectra plots of the local buildings and control room to assure that equipment plots were equal to or more severe than the building plots.

All the specimens successfully completed the seismic testing without exhibiting functional or structural damage.

CONCLUSIONS

A RMS has been designed, built and successfully tested to perform data gathering and control functions in both Class IE and non-Class IE systems in a nuclear generating station. It is believed that this application is the first of its kind to meet Class IE qualification requirements. Full quality assurance compliance with NRC requirements has been achieved by stringent specification and test requirements, which included the necessary functional and seismic tests.

It is hoped that useful design and operating experience will be obtained from this installation and that it will stimulate other application possibilities.

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APPENDIX A

EXPLANATION OF TERMS COMMONLY USED IN RMS

1. Multiplexing - A signaling method using wire path, cable carrier, radio, or combinations of these facilities characterized by the simultaneous and/or sequential transmission and reception of multiple signals in a communication channel including means for positively identifying each such signal.

2. RMS - Remote Multiplexing System. A system in which signals are combined onto the common lines at their point of origin in the field and are separated at the central location.

3. RMP - Remote Multiplexing Panels are mini-multiplexers containing Analog/Digital and Digital/Analog converters, contact and alarm sensing and on/off control inputs and outputs.

4. CCU - Central Control Unit. It polls Remote Multiplexing Panels sequentially and provides data exchange between processors and Control Room Multiplexers.

5. CRM - Control Room Multiplexers which demultiplex data from RMP's and reconstruct them into their field derived equivalents. These units also accept control inputs (analog and on/off) from control boards for transmission to RMP's.

6. DATABUS - which interconnect the CCU to the RMP.

7. SCAN CYCLE TIME - is the time required to update the data in a signal.

REFERENCES

1. A. L. Cava, "Application of A Supervisory Control System," IEEE PES Winter Meeting, 1975.

2. IEEE Standard 323-1971, "Qualifying Class IE Equipment for Nuclear Power Generating Stations."

3. IEEE Standard 344-1971, "Recommended Practices for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations."

4. NRC 10 CFR 50, Appendix B (Quality Assurance Criteria for Nuclear Power Plant and Fuel Reprocessing Plants).

5. Anaconda-I/C Engineering Technical Data Catalog, "Uniplex Data System Model 600 Technical Description."

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ICSB Item 4

CONCERN:

Failure in Vessel Level Sensing Lines Common to Control and Protective Systems

Operating reactor experience indicates that a number of failures have occurred in BWR reactor vessel level reference sensing lines and that in most cases the failures have resulted in erroneously high reactor vessel level indication. For BWR's, common reference sensing lines are used for feedwater control and as the basis for establishing vessel level channel trips for one or more of the protective functions (reactor scram, MSIV closure, RCIC, LPCI, ADS or HPCS initiation). Failures in such sensing lines, may cause reduction in feedwater flow and consequential delay in trip within the related protective channel.

If an additional failure, perhaps of electrical nature, is assumed in a protective channel not dependent on the failed sensing line, protective action may not occur or may be delayed long enough to result in unacceptable consequences. This depends on the logic for combining channel trips to achieve actions.

It is our position that those reference lines common to the feedwater control function and to any of the protective functions for loss of feedwater events be identified, and that the consequences of failures in such reference lines concurrent with the worst additional single failure in the protective systems (reactor scram, MSIV closure, ADS, RCIC, HPCS/HPCI, LPCI, etc.) or their initiation circuits be analyzed.

RESPONSE:

A postulated break in an instrument line plus an additional failure is beyond the design basis for this plant; however, an assessment of plant response to this event is hereby provided.

The instrument reference lines common to feedwater control and to protective system sensors have been identified for this plant. An analysis was performed to determine the consequences of failures in such reference lines concurrent with additional single failures in protective channels not dependent on the failed sensing line. The Sequence of Events are shown in the attachment.

In the highly unlikely scenario, the most severe reference line was assumed to fail such that all attached level instruments erroneously indicated high levels. Then, additional worse-case single failures were postulated in the circuits connected to the remaining 3 reference lines. Worse-case single division power supply loss was considered for ECCS and RCIC, but this is independent from other single failures which could effect RPS or MSIV closure, etc. (i.e., a power bus failure in RPS would fail "safe" causing a trip of that channel). The worst postulated failure path, from the various combinations, was found to be failure of Division IA instrument reference line combined with failure in another component that results in failure of the level 3 reactor scram and isolation function. Worst-case was also assumed for the feedwater controller in that the manual selection switch is on Division IA instrument line and the operator does not take the option to switch control to Division IIA, as he would normally be expected to do when he receives an alarm indicating level instrument mismatch and sees the level mismatch between the indicators. The Feedwater controller responds to the high level error signal by reducing the feedwater flow. When water level decreases to level 4 a low water level alarm is initiated.

When water level decreases to level 3 a second low water level alarm will be initiated but reactor scram and low water level isolation will not occur due to the assumed failures. As water level drop passes through low water level 2 a third low water level alarm will initiate, the HPCS and RCIC system will automatically start and the MSIVs will close. Closure of the MSIVs will result in reactor scram. The water level will continue to drop, but now at a slower rate due to reactor scram and inventory assistance provided by HPCS and RCIC (see attached table "Sequence of Events" and graph of water level vs. time). Assuming the operator still has not switched feedwater to the alternate control (which he would be expected to do), the water level will ultimately reach a minimum level 1.7 feet above the Top of the Active Fuel and then quickly recover to the normal water level range. If the operator does not take action to reduce the HPCS and RCIC flow rate, water level will increase to high level 8 and HPCS and RCIC injection will automatically stop. No fuel failure would occur. The core remains covered at all times. Low pressure systems are also available, but are not necessary because RCIC has more than enough capacity to assure adequate water make-up and inventory control.

The Sequence of Events shown in the attachment shows that the reactor system can withstand any reactor vessel level reference line break coupled with an additional worst single failure in a protective channel not dependent on the failed sensing line without compromising safety. This is assured by the following evaluations:

1. No part of the active fuel is uncovered at any time. This assures no fuel damage and no degradation of the critical power ratio (CPR), or reactivity release.

2. Both the vessel and the containment remain structurally sound throughout the postulated event. This provides secondary assurance that no reactivity can be released to the public.

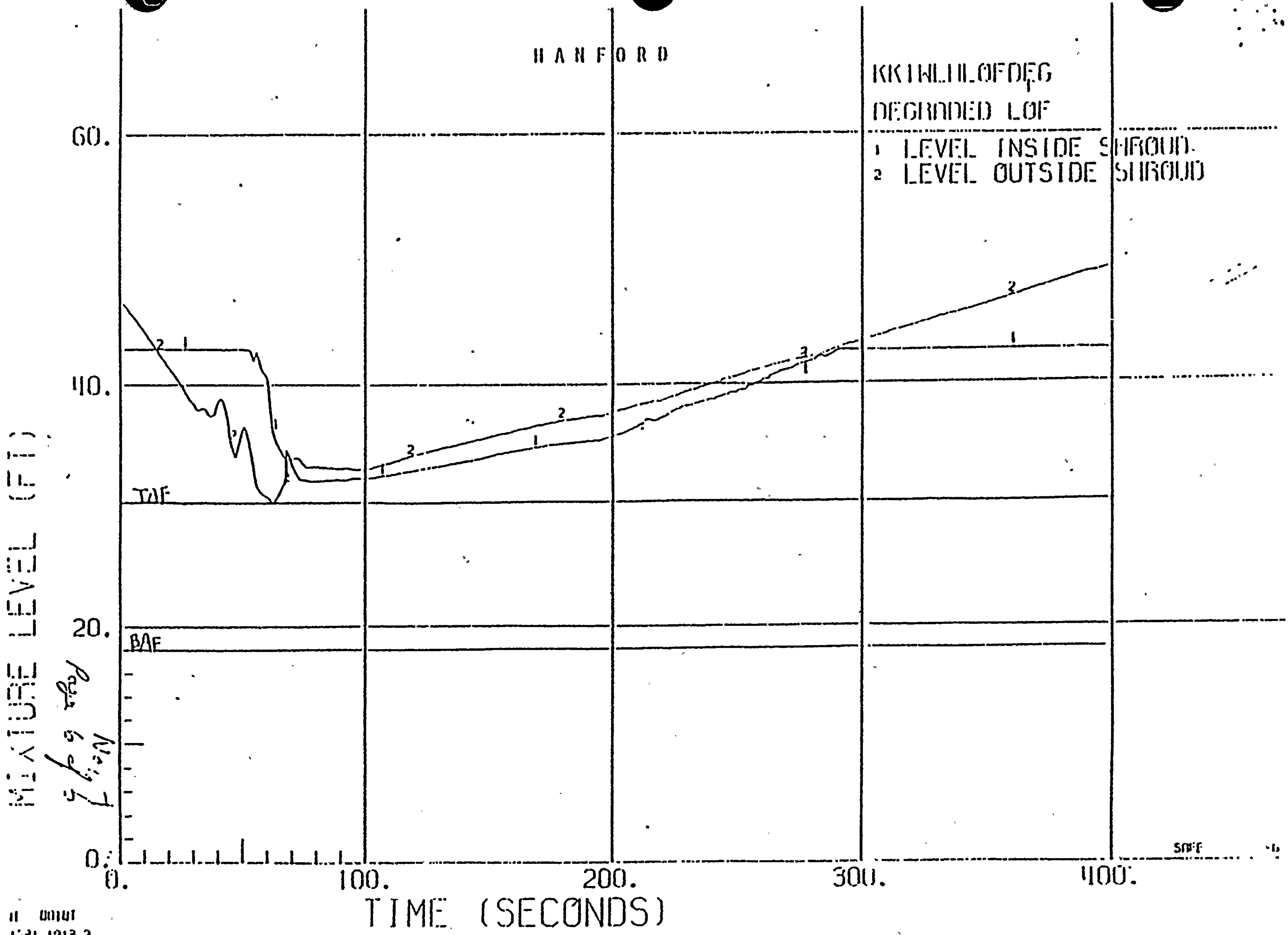
3. The scenario postulated is a highly unlikely event (instrument line breakage with coincident random failure) and compounds it with worst-case conditions throughout the event. Though no credit is taken in this scenario, it is highly probable that the operator would recover feedwater level immediately by switching the controller to the alternate instrument line because of the alarms that call his attention to level indication mismatch and numerous low water level alarms.

There are no failure combinations (i.e., reference line leak/break plus single additional failure) that result in failure of both HPCS and RCIC. One of the two systems are always available. Therefore, failure scenarios that consider failures in the ECCS or RCIC are less limiting (relative to core uncovery) than the failures discussed above.

It is concluded from this assessment of a break in a vessel level sensing line common to control and protective systems plus an additional worst single failure in a protective channel not dependent on the failed sensing line that the resulting accident is less severe and bounded by the DBAs already analyzed in Chapter 15 of the FSAR.

SEQUENCE OF EVENTS

Time (sec)	Events
0	Reactor water level at nominal level
	One of the water level reference legs breaks (assume feedwater control relies on this instrument line).
	Feedwater starts to decrease due to a false high water level reading in the failed instrument line.
3.6	Actual water level drops to L4. No recirculation runback signal due to false reading in the failed channel.
5.0	Feedwater flow decreases to zero.
9.0	Actual water level drops to L3. No low level scram due to the failure of the reference leg and in RPS channel.
31.0	Sensed water level drops to L2 which initiates recirculation pump trip and MSIV closure followed by reactor scram. L2 also initiates HPCS and RCIC.
61.0	HPCS and RCIC flows start to enter vessel.
78.0	Water level reaches minimum and begins to rise quickly. The minimum water level is 1.7 ft above the top of the active fuel.
>78.0	Operator controls water level between L3 and L8 according to level control guideline and brings the reactor to cold shutdown according to cooldown guideline.



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ICSB ITEM 6

Concern:

WNP-2 TMI Item on SRV Position indication - They will modify Appendix B to include use of thermocouples as backup means of detection and emergency procedures for use of them.

Response:

See revised Appendix B, item II.D.3.

Open SER Issues

ICSB-7 Modification of ADS Logic (II.K.3.18), (RSB-9).

ICSB-8a Restart of Core Spray Systems (II.K.21), (RSB-11).

ICSB-8b Separation of HPCI and RCIC System Initiation Levels (II.K.3.13).

Responses to the above issues were submitted in the TMI (Appendix B) sumbittal.

ICSB-9 LPCS and LPCI System Interlocks

A response to this issue was provided in the response to open SER issue 6.3.

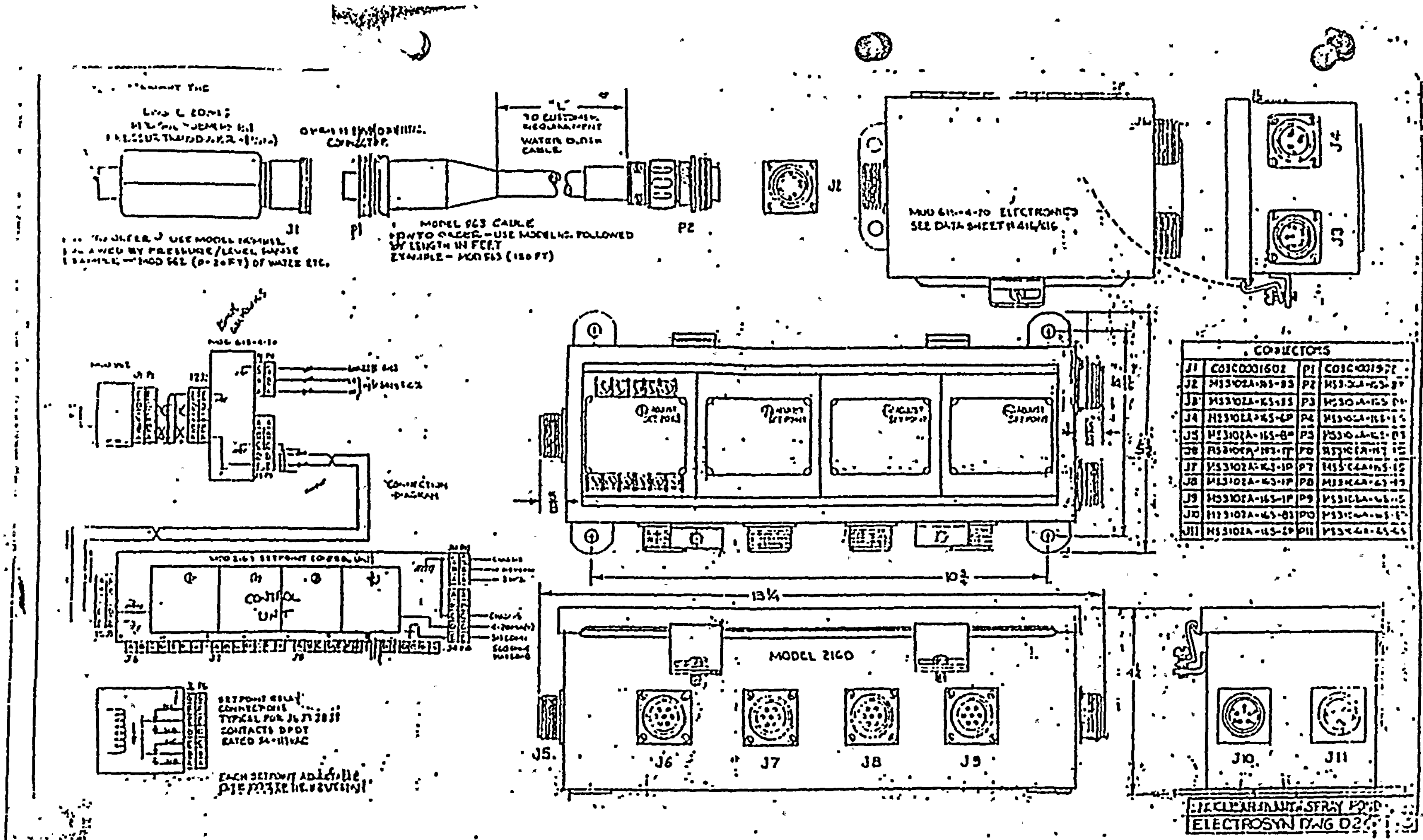
ICSB-Item 10

Concern:

Present design description of the instrumentation and control aspects of the spray pond pumps and the standby service water pumps. Describe how the level instrumentation which trips the SSW pumps on low spray pond level are protected against freezing. (Bring drawings for review).

Response:

Each Standby Service Water Spray Pond level is monitored by a level transducer (Pressure Sensing Probe) located in the pump suction pit about 24' 6" below the spray pond normal water level. Freezing at this depth is not a concern. The transducer sends an electrical signal to a transmitter located on a local instrument rack in the pump house. The transmitter provides the necessary output signals to operate the level switches for pump and valve interlocks and level indication.



ICSB No. 10
Page 2 of 2

ICSB-Item 11

Concern:

FSAR Tables 7.3-3, 5, 7, 23 show level 1 trips at -149 inches. The instrument range extends to -150 inches. Instrument accuracy is stated to be $\pm$7.5 inches. Discuss the effect of errors within the stated accuracy at -149 inches.

Response:

See revised response to NRC Question 031.116, attached.

Q. 031.116
(T7.3-3)
(T7.3-5)
(T7.3-7)
(T7.3-9)

The use of level switches with a range of -150 inches/0/+60 inches to initiate the automatic depressurization system (ACS), the low pressure core spray (LPCS) system and the low pressure coolant injection (LPCI) system with a setpoint of -149 inches as shown in Tables 7.3-3, 7.3-5, and 7.3-7 of the FSAR, respectively, if not a conservative design feature. A similar situation exists for the differential pressure switch on the RCIC turbine steam line where the range is given as -200 inches/0/+200 inches and the high flow trip point is indicated in Table 7.3-9 to be +198 inches. Provide justification for using these instruments whose extreme range is barely above the trip point or the setpoint. Justify the use of these ranges in these applications. Discuss the accuracy of the trip settings and how they are affected by long-term drift and by normal environmental conditions and those occuring during and after postulated accidents.

Response:

The values provided in Chapter 7 are for information only since setpoints are not finialized until the Technical specifications are completed. The actual setpoints for the parameters contained in Tables 7.3-3, 7.3-5, 7.3-7 and 7.3-9 will be shown in the Chapter 16 Technical Specifications. Notes to this effect follow each of the Tables in question.

The values shown on the tables in Chapter 7 will either be updated or deleted and reference made to the Technical Specifications once these specifications are complete.

When the Technical Specifications are issued, the LPCI/LPCS/ADS reactor vessel low water level trip setpoint will be approximately 181 inches.

These set points are derived through application of setpoint margins as delineated in the BWR Standard Technical Specifications taking into account instrumentation drift, loop accuracy, calibration errors, etc.

-129 inches. The RCIC high steam flow trip setpoint will be approximately

ICSB-Item 12

Concern:

Present system description and show how the instrumentation and control aspects of the Main Steam Line Leakage Control System satisfies the single failure criterion (Bring drawings for review).

Response:

Refer to Section 6.7.3.1 (pages 6.7-12, 13 and 14) and response to Q. 031.076 (Amendment No. 14, April 1981).

Open SER Issue

ICSB-A1

Concern:

Provide a discussion as to how the WNP-2 design of the Recirculation Pump Trip (ATWS Interim fix) conforms to Appendix C of NUREG-0460, Volume 3 (Candidate for drawing review meeting December 7-11, 1981).

Response:

Refer to Question 031.115 (attached) and LRG issue RSB-22 in Licensing Review Group submittal (Appendix I).

Q. 031.115
(7.2)

The WNP-2 SER issued at the CP stage of our review in September 1971 acknowledges your commitment to include a recirculation pump trip (RPT) on receipt of a signal indicating high reactor pressure. This trip is intended to mitigate the effects of a failure to scram. Provide the details of your proposed RPT design; indentify and justify any exceptions to the requirements of the reactor protection system (RPS).

Response:

A recirculation pump trip on high reactor pressure is provided to mitigate the effects of failure to scram (ATWS condition). This ATWS RPT is designed to be non-safety related.

A modification of Appendix H to the FSAR provides a description of the equipment and function.*

As stated in the insert to the text, the ATWS RPT does not interact with the RPS, nor are any of the RPS requirements addressed by this function.

*Draft FSAR page change attached.

These functional requirements are satisfied by an elbow flow element where the pressure from the inside to the outside of the elbow is proportional to flow. Consequently, the flow nozzle in the BWR/4 design was replaced by an elbow pressure tap flow element in the recirculation pump suction line for the BWR/5 and /6 design.

H.1.2.8 Recirculation Pump Trip (RPT)

The recirculation pumps are tripped for many reasons, among which are low NPSH, some transients and electrical faults such as short circuits. Only one trip function is currently required to be safety grade, and that function is given the name RPT. The purpose of RPT is to mitigate the thermal consequences of the turbine trip and generator trip transients by tripping the recirculation pumps early in the event, producing rapid pump flow coastdown and additional core voiding, which results in a core reactivity reduction. This system is linked to the reactor protection system (RPS) such that both a scram and a pump trip occur when the turbine stop valves start to close and when turbine governor valve fast closure occurs. Both scram and RPT are bypassed at low thermal power levels.

Since only one power source is available to a BWR/4 pump motor, RPT trips the pumps completely off. The BWR/5 activates the 25% speed source (the low-frequency M-G set) when the pump has coasted down to that speed.

Insert attach

H.1.2.9 Core Flow Measurement

The core flow measurement system is unchanged from the BWR/4 design. For BWR/5 and /6, as an operating convenience, individual jet pump pressure drop signals are fed to the process computer to calibrate the system and obtain the jet pump integrity surveillance data required by the technical specifications.

H.1.2.10 Recirculation System Operation

Due to the changes described in Subsections H.1.2.2 and H.1.2.4, the startup and operation of the BWR/5 recirculation system is significantly changed from previous systems. As a result, new control interlocks were necessary to prevent significant transients, equipment damage, or unnecessary scrams. Electrical interlocks were installed between the LFMG set and the normal power supply to prevent damage to the LFMG set and on the flow control valve to prevent cavitation damage. These interlocks also protect against flow-increase transients when starting the system or transferring to the normal power supply.

Insert to Page H.1.2-4:

In addition to the RPT associated with a reactor scram, and the normal pump trips, a high vessel pressure or low low vessel level (level 2) will initiate a recirculation pump motor trip without transfer to the 25% speed source. Each trip sensor and channel is separate and independent from the reactor protection system, and includes a testability feature that will allow testing of each trip sensor while the recirculation system is in operation. The abnormal position of the test switch is annunciated.

ICSB-Item A3

Concern:

Describe any design change to the scram discharge volume level sensing system? Diversity? Present design is float switches. Are you going to have the standard GE design of ΔP level sensing and float switch?

Response:

See response to NRC Question 010.41 attached. Standard GE design of ΔP level sensing and float switch will be provided at WNP-2.

Q. 010.041
(4.6)

Demonstrate that the scram discharge system meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980.

Response:

The scram discharge system for WNP-2 has been evaluated against the Generic Safety Evaluation Report, "BWR Scram Discharge System", dated December 1, 1980. In short, the evaluation indicated that the WNP-2 scram discharge system needed upgrading in the following areas:

1) Addition of redundant vent and drain isolation valves;
2) Addition of redundant and diverse level instrumentation for scram;
3) Relocation and repiping of instrument piping directly to the scram instrument volume;
4) Addition of new surveillance and operating procedures.

A summary of our evaluation results is provided below:

FUNCTIONAL CRITERIA

1. The scram discharge volume (SDV) shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control-rod-drive scram performance.

   WNP-2 Compliance:

   WNP-2's SDV system is currently designed to meet the 3.34 gallons per drive requirement specified in the GE Design Specification 22A4260. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SAFETY CRITERIA

1. No single active failure of a component or service function shall prevent a reactor scram, under the most degraded conditions that are operationally accepted.

   WNP-2 Compliance:

   The WNP-2 system has been designed to meet single failure criteria. The SDV is designed with an integral instrument volume (IV) which provides direct and immediate detection of liquid accumulation. The SDV instrumentation is redundant and single failure proof (including partial loss of service functions).

2. No single failure shall result in uncontrolled loss of reactor coolant.

WNP-2 Compliance:

A redundant air-operated vent valve and drain valve will be added on the SDV in series to insure system isolation during reactor scrams. This includes independent solenoid valves for each set of air-operated vent and drain valves.

3. The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

WNP-2 Compliance:

Six additional diverse level sensors will be added to the SDV system to ensure diversity and redundancy in level monitoring and scram functions. Common cause failures will be considered in the selection of the instruments. This is in agreement with Alternative 3 of the "Acceptable Compliance" statement for this item in the Generic SER.

4. System operating conditions which are required for scram shall be continuously monitored.

WNP-2 Compliance:

The addition of the level switches described in 3 above and periodic surveillance testing of the instruments will provide a continuous means of monitoring the SDV liquid level and insuring instrument reliability. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

WNP-2 Compliance:

During routine surveillance testing, instrument repair or calibration the associated logic will be placed in a half-scram (1 out of 2) configuration in accordance with the plants technical specifications.[1] This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

OPERATIONAL CRITERIA

1. Level instrumentation shall be designed to be maintained, tested, or calibration during plant operation without causing a scram;

2. The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation;

3. The system shall be designed to minimize the exposure of operating personnel to radiation;

4. Vent paths shall be provided to assure adequate draining in preparation for scram reset;

5. Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

WNP-2 Compliance:

1. The system logic is designed as a one out of two, twice configuration. Each of the associated instrument channels is capable of being separately isolated for maintenance, testing or calibration without inadvertently scramming the reactor.

2. The SDV is provided with a high liquid level alarm on each IV to alert the operator to liquid accumulation in the SDV.

3. The SDV system has been designed in accordance with GE design specification 22A4260 to minimize the exposure of operating personnel to radiation. In addition, the system is being reviewed as part of the WNP-2 ALARA program.

4. The SDV vents directly to the reactor building atmosphere and is independent from other plant vent system.

5. The vent and drain system for the SDV is totally independent from other plant systems, and is therefore not susceptible to blockage or water buildup through system interfaces.

DESIGN CRITERIA

1. The scram discharge headers shall be sized in accordance with GE OER-54 and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on a plant-specific maximum inleakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum inleakage is the maximum flow rate through the scram discharge line without control-rod motion summed over all control rods. The analysis should show no need for vents or drains.

   WNP-2 Compliance:

   WNP-2's IVs have been designed as vertical extensions attached directly to the SDV. This configuration provides a direct hydraulic couple between the SDV and IVs and insures immediate and continuous liquid level monitor in the SDV. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

2. Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

   WNP-2 Compliance:

   WNP-2's SDV is adequately coupled to the IV to allow proper instrument operation. The SDV instrument set-point for scram was established to insure an available volume of 3.34 gallons per drive (185 drives). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

3. Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

   WNP-2 Compliance:

   All the WNP-2 SDV instrumentation will be relocated and repiped directly to the IV instead of the vent and drain piping. Procedures will be modified to include functional testing of SDV level instrumentation after each scram. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

4. The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

WNP-2 Compliance:

The addition of the redundant and diverse instruments described under Safety Criterion 3 and rerouting of the instrument piping to the IV provide an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

5. Structural and component design shall consider loads and conditions including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

WNP-2 Compliance:

The WNP-2 SDV design compliance with the latest GE design criteria as outlined in GE Design Specification 22A4260. In addition, the system will be reviewed as part of the equipment qualification program.

6. The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

WNP-2 Compliance:

WNP-2's present design configuration meets these requirements.

7. Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

WNP-2 Compliance:

WNP-2's SDV header system is designed as a continually expanding path from the 185 3/4" individual scram discharge (withdrawal) lines to one of two integrated SDV/IV systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting

of seven 6" return headers from the individual hydraulic control unit (HCU) banks to an 8" combined return header) to the 12" vertically oriented IV. The location where blockage need be assumed (piping less than 2" diameter) is in the 3/4" discharge line from the individual HCU. Blockage here would only cause failure of one control rod to insert. This is an acceptable consequence for a single failure and has been evaluated as part of the plant design basis. Accordingly, this design complies with the "Acceptable Compliance" statement for this item in the Generic SER.

8. System piping geometry (i.e., pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation.

   WNP-2 Compliance:

   The WNP-2 SDV has been designed to insure a positive downward slope of scram header and drain piping.

9. Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

   WNP-2 Compliance:

   Each IV is provided with high liquid level and rod block instrumentation attached directly to it. The generic SER states that this is acceptable.

10. Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

   WNP-2 Compliance:

   As stated under Safety Criterion 2, redundant air-operated vent and drain valves will be provided for system isolation. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SURVEILLANCE CRITERIA

1. Vent and drain valves shall be periodically tested.

WNP-2 Compliance:

The vent and drain valves will be tested in accordance with the plant technical specification to verify valve closure[1] in less than 30 seconds (current GE specification). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

2. Verifying and level detection instrumentation shall be periodically tested in place.

WNP-2 Compliance:

The SDV instrumentation will be tested in accordance with the plants technical specification which will include post scram testing to verify instrument operability.[1]

3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control-rod density.

WNP-2 Compliance:

Surveillance testing will be performed in accordance with the plants technical specifications.

[1] The plant technical specifications will be based on the Standard Technical Specifications for Boiling Water Reactors, NUREG-0123, provided by the NRC.

ICSB-Item A5

Concern:

Drawing Review - HPCS and RCIC automatic switchover of suction source from condensate storage. Show that separation exists between HPCS and RCIC sensors for the transfer and describe why seismic events do not affect the transfer. Describe any manual valve interlocks and how controlled.

Response:

See revised responses to NRC Questions 031.128, 211.146 and 211.197 (attached).

Q. 031.128
(7.3.1)

Question:

Figure 7.3-7 of the FSAR indicates that there are two condensate storage tanks, each with a manually operated discharge valve. The functional control diagram (i.e., Figure 7.3-8) illustrates, and the text discusses, the interlock between the condensate storage tank and the suppression pool suction valves which is intended to provide assurance that the high pressure core spray (HPCS) system pump has an acceptable supply of water at the suction inlet. However, if both manual discharge valves were to be closed, the purpose of this interlock would be defeated. Accordingly, provide justification in Section 7.3.1.1.1.1 for the omission of manual discharge valve position switches as initiators in the HPCS pump suction control logic.

Response:

FSAR Figure 7.3-7 has been changed to Figure 6.3-1 and Figure 7.3-8 has been changed to Figures 7.3-8a, 7.3-8b, and 7.3-8c. These figures have been revised to delete the manual discharge valve position interlocks. These interlocks were originally provided (when the condensate storage tanks level switches were physically mounted on the tanks) to prevent premature transfer of the HPCS suction to the suppression pool when one of the tanks was out of service (drained) giving a false transfer signal.

Figure 6.3-1 has been revised to indicate the level switches on a Class 1 standpipe in the Reactor Building on the condensate supply line to HPCS which is downstream of the manual suction valves. (These valves are locked open during normal operation). With this modification, valve position interlocks are no longer required. FSAR Section 7.3.1.1.1.1 has been revised to correctly describe this modification.

b. Automatic depressurization system (ADS);

c. Low pressure core spray system (LPCS);

d. Low pressure coolant injection (LPCI) mode of the residual heat removal system (RHRS).

The following plant variables are monitored and provide automatic initiation of the ECCS when these variables exceed predetermined limits:

1. Reactor Vessel Water Level

A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the reactor coolant pressure boundary and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes. Refer to Figure 7.3-9 (Nuclear Boiler P&ID) for a schematic arrangement of reactor vessel instrumentation.

2. Drywell Pressure

High pressure in the drywell could indicate a breach of the reactor coolant pressure boundary inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes.

7.3.1.1.1.1 High Pressure Core Spray (HPCS) System - Instrumentation and Controls

a. HPCS Function

The purpose of the HPCS is to provide high pressure reactor vessel core spray for small line breaks which do not depressurize the reactor vessel. In addition, HPCS is redundant to the RCIC system for mitigation of the consequences of various events listed in Appendix 15A. Refer also to 6.3.2.2.1.

b. HPCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure ~~7.3-7~~ 6.3-1 (HPCS P&ID). HPCS system component control logic is shown in Figure 7.3-8 (HPCS FCD) and Figure 7.3-4 (HPCS Power Supply FCD). Instrument specifications are listed in Tables 7.3-1 and 7.3-2. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure ~~7.3-7~~ 6.3-1 (HPCS P&ID) and Figure 7.3-8 (HPCS and HPCS Power Supply ~~FCD~~ FCDs).

Revise for SCN ~~81-435~~ Q 31.128

mounted on a Class 1 standpipe in the Reactor Building
closes. Two level switches are used to detect low water level in ~~each of~~ the condensate storage tanks. Either switch can cause automatic suction transfer. The suppression pool suction valve also automatically opens if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level and either switch can initiate opening of the suppression pool suction valve. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other closes.

The HPCS provides makeup water to the reactor until the vessel water level reaches the high level trip (Trip Level 8) at which time the injection valve MOF004 is automatically closed. The pump will continue to run on minimum flow recirculation. The injection valve will automatically reopen if vessel level again drops to the low level (Trip Level 2) initiation point.

The HPCS pump motor and injection valve are provided with manual override controls. These controls permit the reactor operator to manually control the system following automatic initiation.

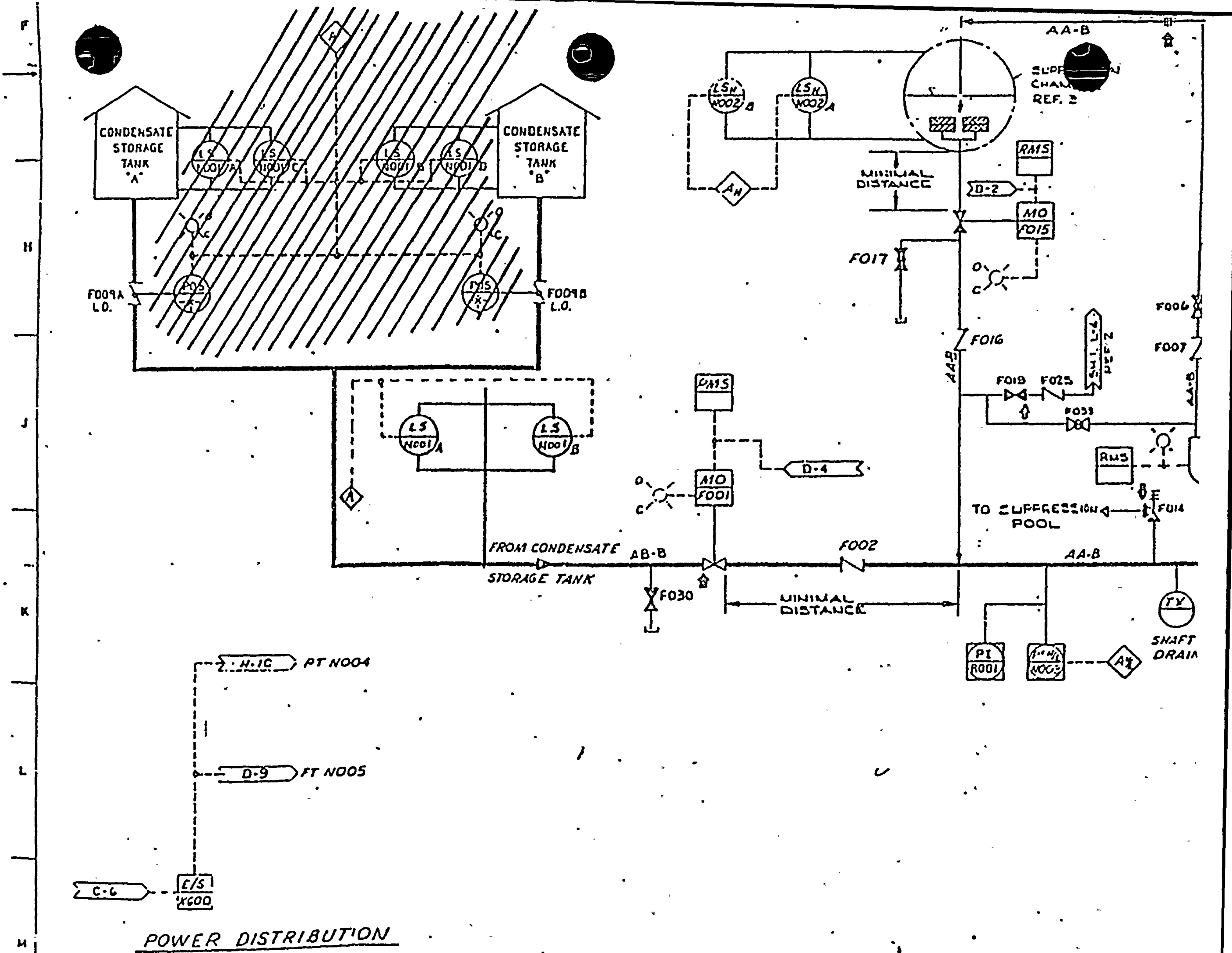
7.3.1.1.1.2 Automatic Depressurization System (ADS)-Instrumentation and Controls

a. ADS System Function

The automatic depressurization system is designed to provide automatic depressurization of the reactor vessel by activating seven safety/relief valves. These valves vent steam to the suppression pool in the event that the HPCS cannot maintain the reactor water level following a LOCA. ADS reduces the reactor pressure so that flow from the low pressure ECCS, LPCI system and LPCS, can inject into the reactor vessel in time to cool the core and limit fuel cladding temperature. Refer also to 6.3.2.2.2.

b. ADS Operation

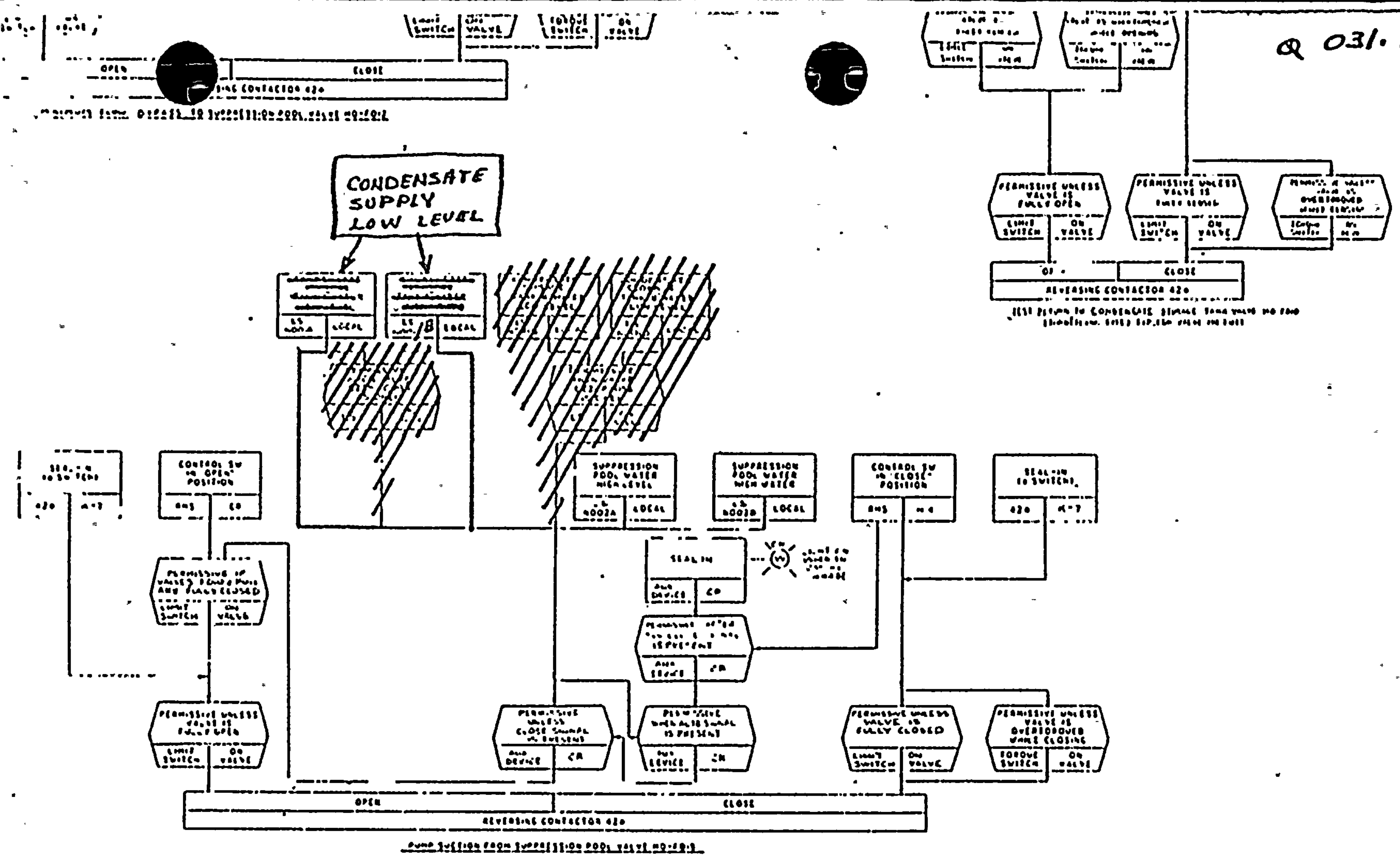
Schematic arrangements of system mechanical equipment is shown in Figure 7.3-9 (Nuclear Boiler P&ID). ADS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD). Instrumentation specifications are listed in Tables 7.3-3 and 7.3-4. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-9 (Nuclear Boiler P&ID) and Figure 7.3-10 (Nuclear Boiler FCD).



HPCS P&ID

Revise for BRSCN 81-435

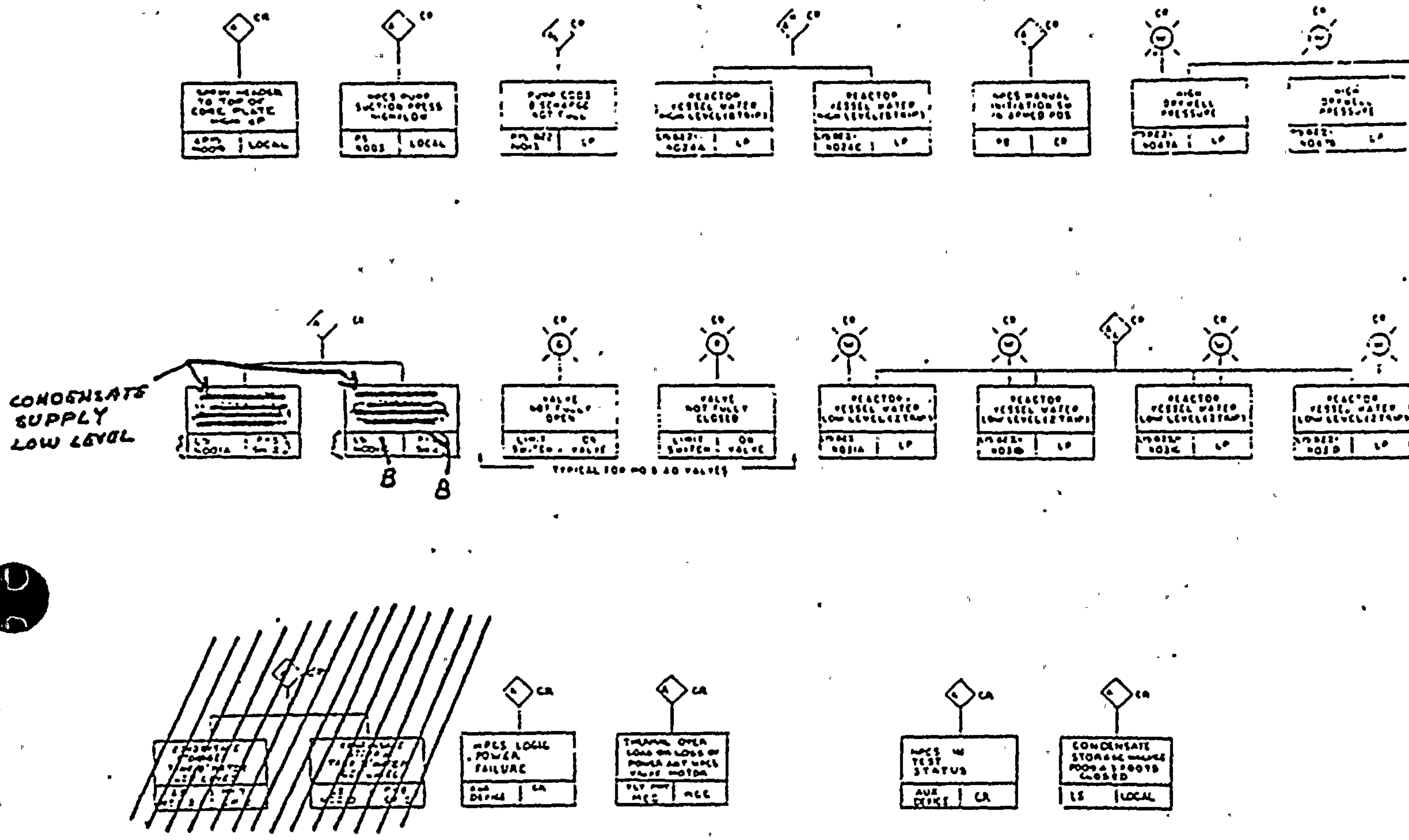
Q 031.128



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRA FCD SHEET 2

Figure 7.

Q031.128



HPCS FCD Figure

WASHI

Q. 211.146
(5.4.6)

Question:

In the responses to Questions 211.046 and 031.015, it is stated that an automatic safety-grade switchover from the condensate storage tank to a Seismic Category I supply (i.e., the suppression pool) has been provided as a convenience to the operator. Provide a description of the automatic switchover feature and its initiating signal and confirm that both electrical and mechanical features are safety grade.

Response:

The automatic switchover feature for HPCS and RCIC consists of two Class IE level switches for each system which will be mounted on a standpipe in the pump suction line. This standpipe is located on the common condensate supply line inside the Reactor Building at the Reactor Building/Service Building interface.

The standpipe is open ended and is used to indicate either a low water level condition in the Condensate Storage Tanks (CST) or a loss of suction supply from the CST. The standpipe is designed, fabricated and installed to Seismic Category 1, Quality Class 1 and ASME Section III Class 2 standards.

The piping from the Reactor Building/Service Building interface to both the RCIC and HPCS systems have been upgraded to Seismic Category I; each circumferential buttweld has been radiographically examined per ASME Section III, NC-5230, and a chemical analysis has been performed on all piping materials and as-deposited weld materials.

The HPCS P&ID (Figure 6.3-1) and Functional Control Diagram (FCD (Figure 7.3-8) and the RCIC P&ID (Figure 5.4-9) and FCD (Figure 7.4-2) have been revised to indicate this design feature.

Revise for SCN 81-435
Q 211.146
PTM 9/11/81

where the shutdown coolant system can be placed into operation.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the make-up water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat.

Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine driven pump will supply demineralized make-up water from the condensate storage tank to the reactor vessel. The suction line from this source is provided with an in-line reserve ~~tank~~ with appropriate safety-related level instrumentation. In the event that the water supply from the condensate storage tank becomes exhausted, the level instrumentation in the in-line reserve ~~tank~~ initiates an automatic switchover to the suppression pool as the water source for the RCIC pump. The in-line reserve ~~tank~~ has sufficient volume to maintain the minimum required RCIC pump NPSH plus a two foot margin while the switchover occurs, thus assuring a water supply for continuous operation of the RCIC system. The turbine will be driven with a portion of the decay heat steam from the reactor vessel, and will exhaust to the suppression pool.

During RCIC operation, the suppression pool shall act as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in the Residual Heat Removal System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly or by condensing generated steam prior to entering the suppression pool. When using the steam condensing mode, the condensate discharge from the heat exchangers may be used as RCIC pump suction supply.

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC Systems.

a. A schematic "Piping and Instrumentation Diagram" (Figure 5.4-9) shows all components, piping, points where interface system and subsystems tie

Q. 211.012
(4.6)
(5.4.6)
(5.4.7)

Revise for SCN 81-455
Q 211.146
PTM 9/11/81

Describe the provisions incorporated into the WNP-2 facility to protect the RCIC and the RHR systems from cold weather and from dust storms and to assure satisfactory operational performance under any adverse meteorological conditions. In this discussion, include consideration of the standby liquid control system and the control rod drive (CRD) hydraulic system and any other sources of water for these systems (e.g., the condensate storage tank and the standby service water).

Response:

The RCIC system takes suction from the condensate storage tanks during normal modes of operation. The condensate storage tanks are provided with heaters to maintain water temperature above 40°F at all times. All above ground piping that contains water is heat traced to prevent freezing. Since the CST is a covered tank, the water supply is not affected by dust storms. To provide a Category I source of cooling water for the RCIC system, ~~an alternate path of cooling water can be valved in from~~ the suppression pool, which is inside the reactor building and protected from cold weather and dust storms.

automatic transfer circuitry has been provided to transfer suction from the CST

The control rod drive hydraulic system normally takes suction from the main condensate system, downstream of the condensate demineralizers. All the piping is located within the Turbine Building or Reactor Building. The secondary source of water is the condensate storage tank if the main condensate system is not available. Both sources of water are protected from cold weather and dust storms.

The standby liquid control system, which is filled with sodium pentaborate, is provided with tank heaters and heat tracing to prevent solidification. The entire system is located within the Reactor Building, so it is unaffected by cold weather or dust storms.

The RHR system takes suction from either the recirculation piping or the suppression pool. All the piping is within the Reactor Building.

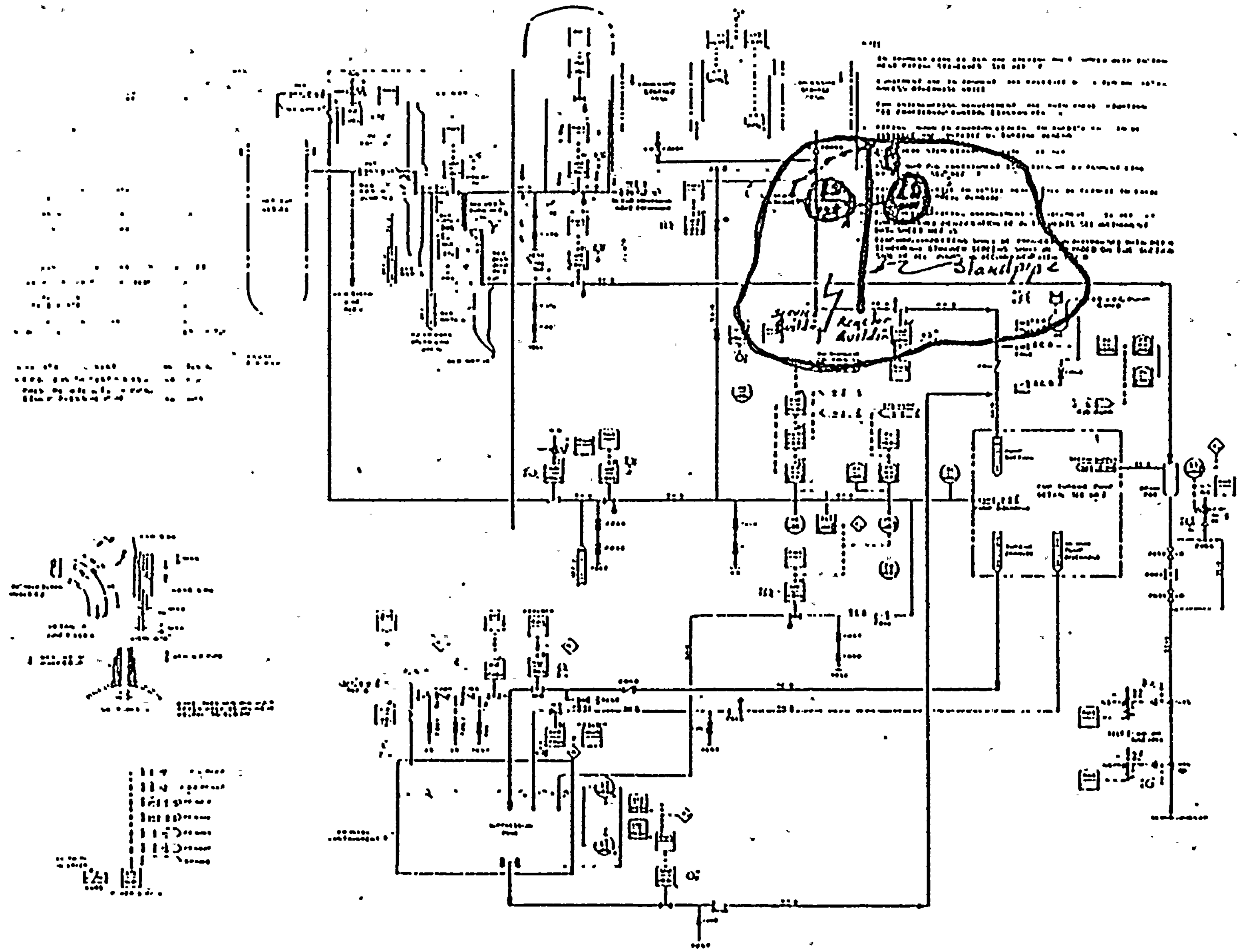
The RHR heat exchangers dissipate their heat to the standby service water system. All SW piping and components are either below the frost line, within the heated pumphouse, or, in the case of the spray rings, kept drained by the return header drain valve when not in operation. The SW pump suction is 26

Q. 211.099
(7.5)

Since systems such as the HPCS, HPCI, and RCIC are initially aligned to draw coolant water from the CST and switch to the suppression pool following a signal indicating a low water level in the CST, it is our position that the CST water level should be included in Table 7.5-1 of the FSAR, entitled "Safety-Related Display Instrumentation." Accordingly, add the signal indicating low water level in the CST in Table 7.5-1. Alternatively, justify its omission.

Response:

WNP-2 design includes an indication of condensate storage tank level in the control room meeting the requirements of Regulatory Guide 1.97 Rev. 2. This indication will be described in Section 7.5 and included in Table 7.5-1 when this section is amended to discuss the requirements of Regulatory Guide 1.97.

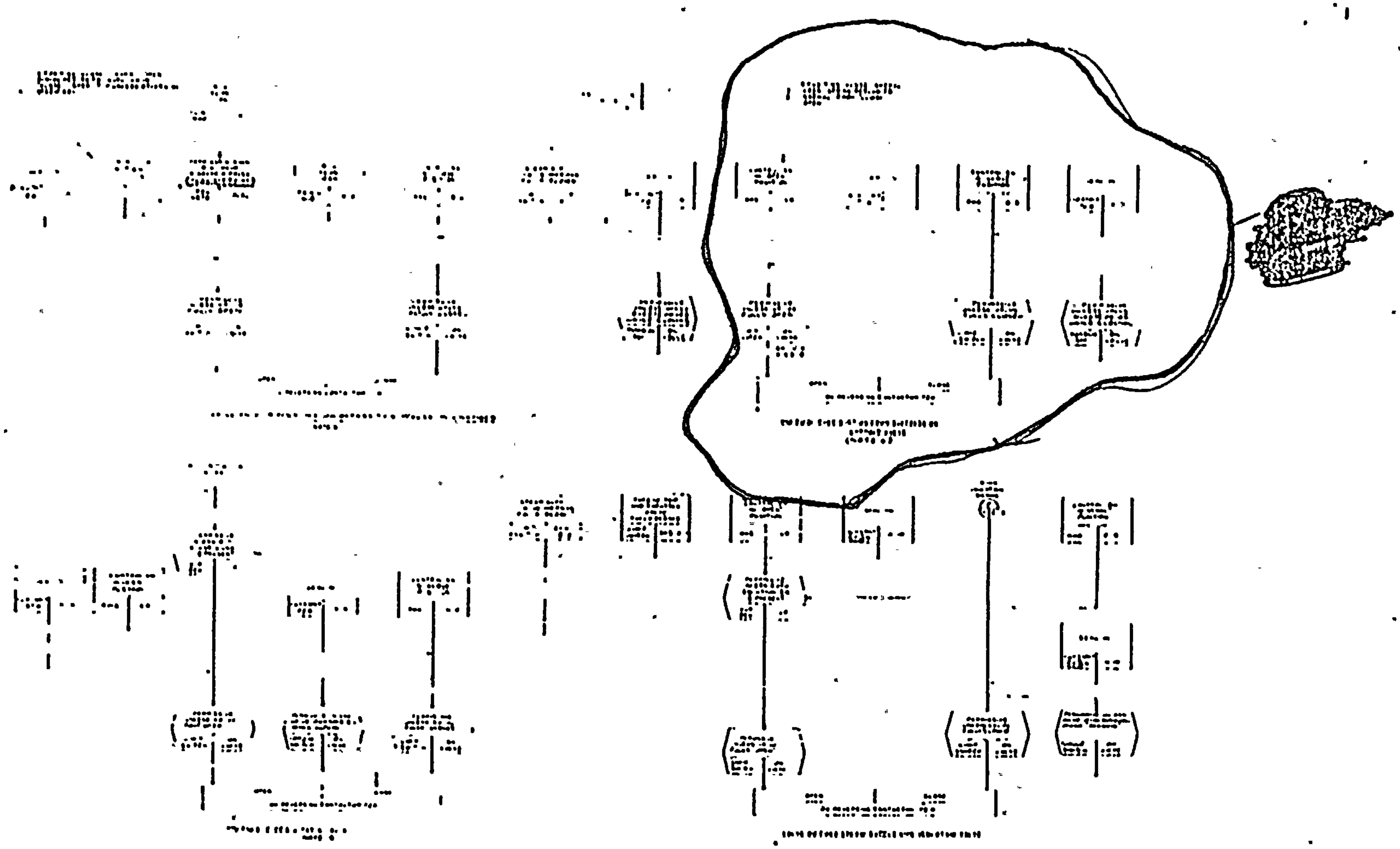


Revise For

81-435
Q211.146

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	P & ID RCIC SYSTEM	FIGU 5.4-

AMENDMENT NO. 16
June 1981

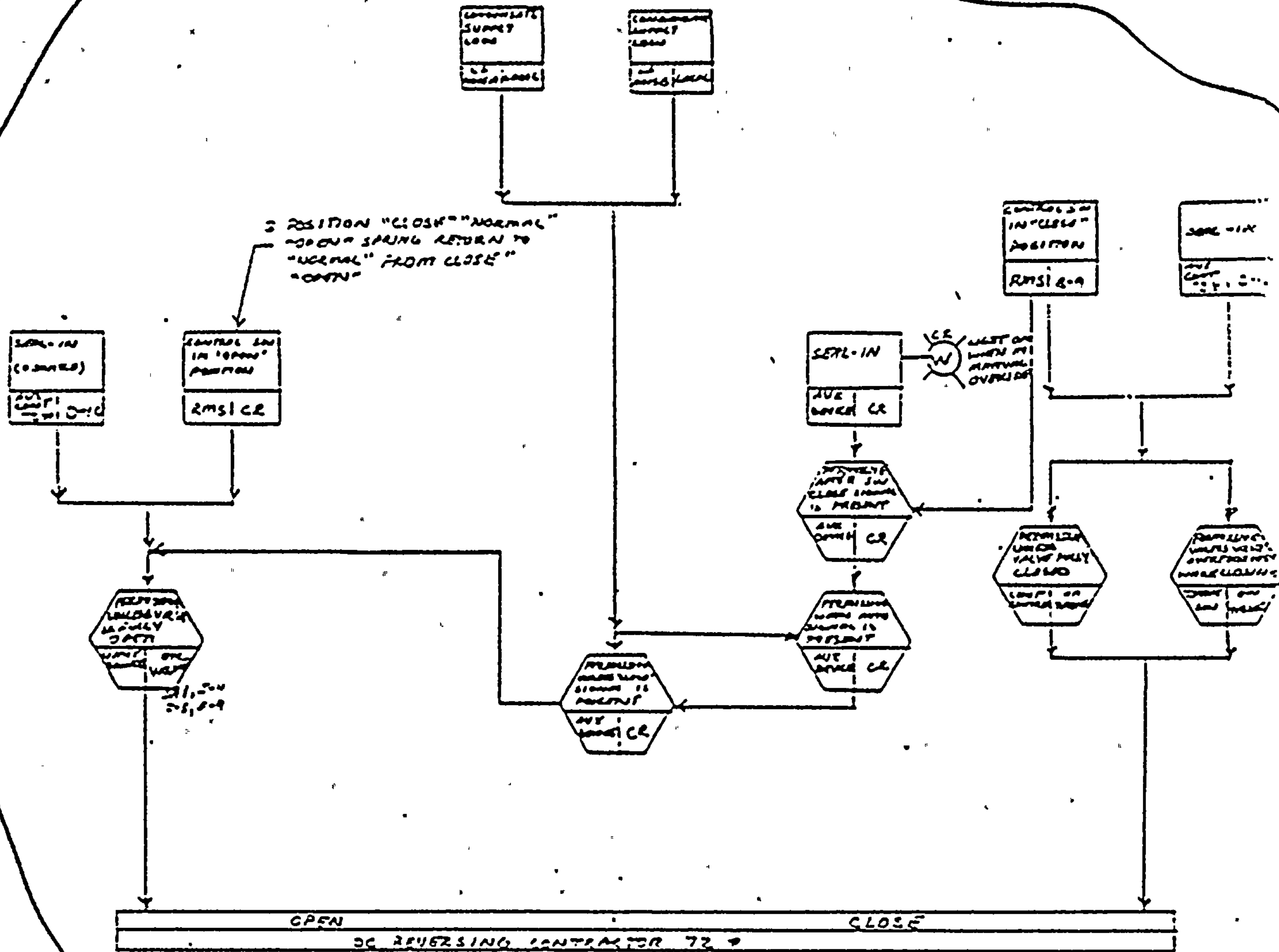


Revise Per
BRSCN 81-435
Q 211.146

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RCIC SYSTEM FCD	FIGURE 7.4-21

REVISE TO BRSCN 81-435
Q 211.146

Add to Figure 7.4-2b



PUMP SUCTION FROM SUPPRESSION POOL VALVE MO-F031
(NOTE 3)

Q. 211.197

Question:

Section 6.3.2.2.1 of the FSAR states that the HPCS system will automatically switch over from the condensate storage tank (CST) to the suppression pool if the CST water supply becomes exhausted or is not available. Review of Figure 7.3-10b indicates that automatic switchover will only occur if the CST water level drops to the minimum level and activates any one of the four level switches (two per tank). However, in the event that CST water cannot be supplied to the pump while the CST water level is above the minimum water level, automatic switchover is precluded. Resolve this apparent discrepancy between the P&IDs and Section 6.3.2.2.1.

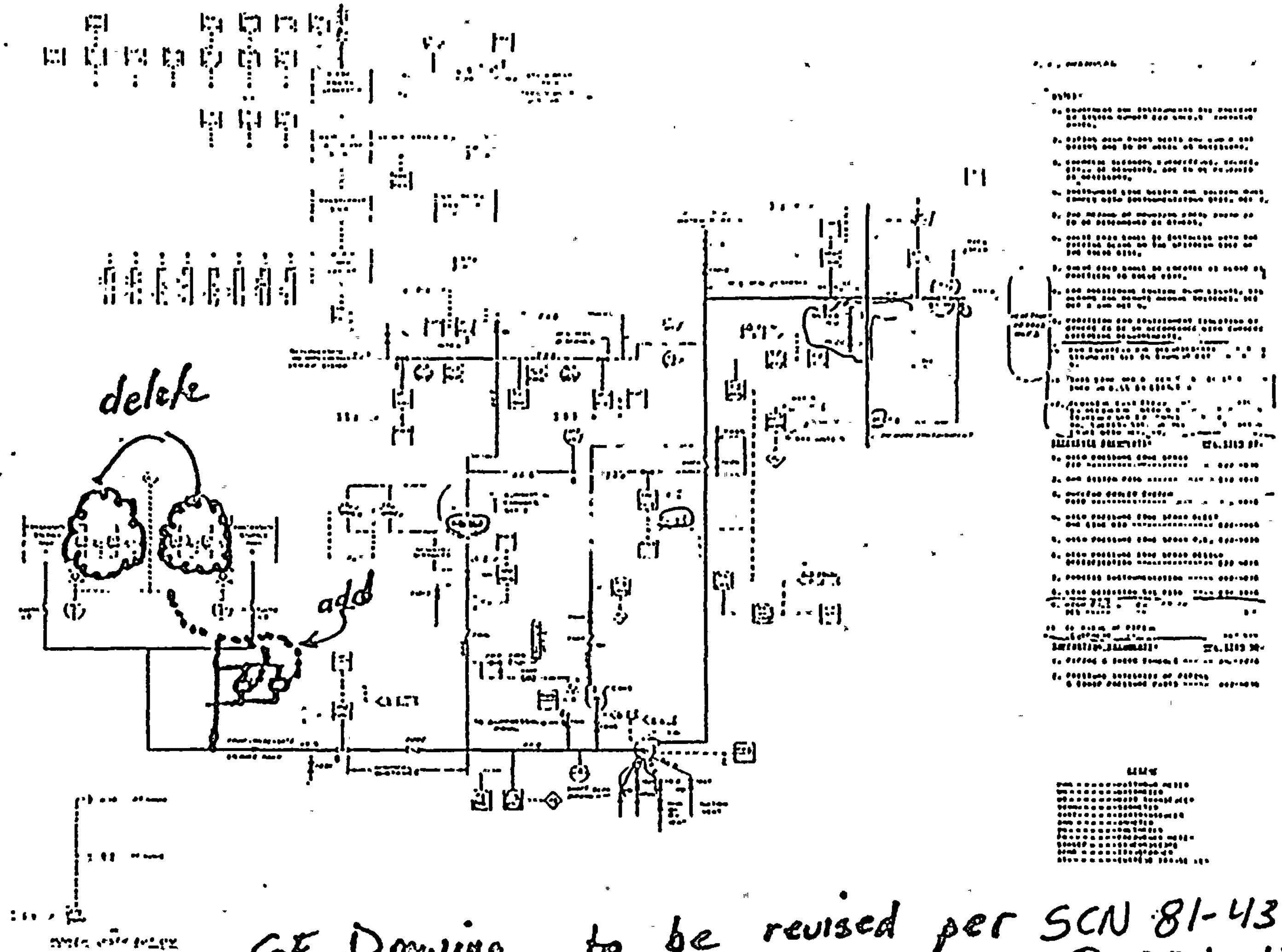
Response:

Figure 7.3-10b, High Pressure Core Spray, Functional Control Diagram Sheet 2, has been changed to Figure 7.3-8b in Amendment 10. The level indicators which provide the signal for automatic switchover of both HPCS and RCIC are mounted on a Seismic Category I standpipe in the Reactor Building. These level indicators as installed will sense a loss of suction supply as well as low level in the condensate storage tanks for the non-Seismic Category I portion of the condensate system. The piping downstream of the standpipe has been upgraded to Seismic Category I and will guarantee a suction supply during suction switchover to the suppression pool. Figure 6.3-1, HPCS P&ID has been revised to indicate these changes. See also the revised response to Question 211.146.

BESCN 81-435
Q 211.197

For Info Only

AMENDMENT No. 16
June 1981



For Information BRSCN 81-435, Q211.197

10/19/81

GE Drawing to be revised per SCN 81-435
Q 031.128

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HPCS P&ID	FIGURE 6.3-1

PSB-E-misc D.G Loading Sequence

REMARKS: As agreed to in the meeting, the statement on p 8.3-2 has been clarified in the revised Ch. 8.3.

RESOLUTION: The Supply System has submitted a simplified logic diagram describing the loading sequence of the diesel generator (including starting and operation and the second level of undervoltage protection). This information will be included in the FSAR. The Supply System also agrees to reflect the use of electro-mechanical relays in diesel Generator load sequencing on p 8.3-2 of the revised ch. 8.3.

PSB - ELECTRICAL

ITEM 8.3 REVISION

REMARKS: As agreed to in the attached correspondence, a revised Ch. 8.3 has been submitted to the NRC informally. This change will be included in the FSAR in a future amendment.

RESOLUTION:

Revised Chapter 8.3 submitted January 13, 1982, in G02-82-31.

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

January 13, 1982
G02-82-31
SS-L-02-CDT-82-011

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555.

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SECTION 8.3 REWRITE

Reference: Letter, R. L. Tedesco to R. L. Ferguson, "WNP-2 - Request for Additional Information", dated July 22, 1981

Enclosed are sixty (60) copies of the draft revised pages for the WNP-2 FSAR Section 8.3 rewrite. Included in this enclosure is the response to NRC Question 040.088 which was transmitted to the Supply System by the reference letter.

Responses to the remaining Power Systems Branch Questions will be submitted to the NRC prior to January 31, 1982 as committed to in the PSB meeting December 16-18, 1981. Both the Section 8.3 rewrite and the NRC question will be included in Amendment 23.

Very truly yours,

G. D. Bouchey, Deputy Director
Safety & Security

CDT/ct
Enclosure

cc: R. Auluck - NRC
WS Chin - BPA
R. Feil - NRC-Site

8201260266

DRAFT SER OPEN ITEMS

- o Draft SER of 11/17/81
- o Draft SER inputs from SEB (11/18/81), RSB (11/27/81), CPS (12/1/81), PSL (11/6/81), CMEB (telecon)

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
2.3.3(p.4)	Emergency Preparedness Plan Meteorological Requirements	CLOSED	* (1)
2.5.4.3/ 2.5.4.5	Soil Backfill Compaction and Testing Review (Geotech # 1 through 8)	CLOSED	G02-81-527 12/15/81
3.5.1.1	Internally Generated Missiles Outside Containment (ASB 3.5.1.1)	OPEN	01/31/82
3.5.1.2	Internally Generated Missiles Inside Containment (ASB 3.5.1.2)	OPEN	01/31/82
3.5.2	Tornado Utility Pole Missile Requirements (ASB 3.5.2)	CLOSED	*
3.6.2.a	Break Location Stress Data Summary (MEB-1)	CLOSED	N.L.U. (2)
3.6.2.b	Postulation of Breaks (MEB-2)	CLOSED	*
3.6.2.c	Correlation of Class I Stress Allowables (MEB-3)	CLOSED	*
3.6.2.d	Correlation of Class I Stress Allowables (MEB-4)	CLOSED	*
3.6.2.e	Break Opening Time (MEB-5)	CLOSED	N.L.U.
3.6.2.f	Clarification of 3.6.2.5.4.11.c (MEB-6)	CLOSED	*
3.7.1	Radwaste Building Analysis: Q 130.055, 130.056 (SEB-2, -17)	OPEN	1/31/82
3.7.3(a)	Radwaste Building SSI Analyses Comparison	OPEN	1/31/82
3.7.3(b)	Spray Pond Analysis: retaining wall conservatism (SEB-32)	CLOSED	G02-81-518 12/14/81
3.7.3.a	Reduced Load Factor Justification (MEB-7)	CLOSED	*
3.7.3.b	Clarification of 3.7.3.2.1 (MEB-8)	CLOSED	*
3.7.3.c	Justification of 3.7.3.2.2 (MEB-9)	CLOSED	*

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
3.8.2	Steel Containment Ultimate Capacity Analysis (SEB-1)	CLOSED	G02-81-518 12/14/81
3.8.2(a)	Containment Shell Fatigue Analysis	CLOSED	"
3.8.2(b)	Effects of Shell Stiffening and Opening	CLOSED	"
3.8.2(c)	Design of Concrete Above/Below Bottom Head	CLOSED	"
3.8.2(d)	NUREG 0808 Assessment (DAR Rev. 3)	OPEN	DAR (3)
3.8.3(a)	Internal Structure Compliance with ACI-349 Code/RG 1.142	CLOSED	G02-81-518 12/14/81
3.8.3(b)	Basemat Compliance with ASME Code, Section III, Division 2	CLOSED	"
3.8.4(a)	Cat I Structures' Compliance with ACI-349 Code/RG 1.142 (SEB-11)	CLOSED	"
3.8.4(b)	Design and Analysis of Spent Fuel Pool Structures; 130.076	CLOSED	*
3.8.5(a)	Reactor Building Foundation Compliance with ASME Code, Section III, Division 2	CLOSED	G02-81-518 12/14/81
3.8.5(b)	Other Cat I Foundations' Compliance with ACI-349 Code/RG1.142	CLOSED	"
3.9.1.a	Seismic Transient for Components (MEB-10)	CLOSED	*
3.9.1.b	Correct Errors in Table 3.7.4 (MEB-11)	CLOSED	*
3.9.1.c	Complete Table 3.9-15 (MEB-12)	CLOSED	*
3.9.1.d	Verification of Computer Codes (MEB-13)	CLOSED	*
3.9.1.e	ECCS Pump Motor Rotor (MEB-15)	CLOSED	*
3.9.1.f	Orificed Fuel Support Stress Analysis (MEB-16)	CLOSED	*
3.9.1.g	Bracing of Hydraulic Control Units (MEB-17)	CLOSED	*
3.9.2.1.a	Piping Restraint Installation (MEB-18)	CLOSED	*
3.9.2.1.b	Thermal Expansion Effects (MEB-19)	CLOSED	*

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
3.9.2.1.c	Steady State Pipe Vibration (MEB-20)	CLOSED	*
3.9.2.1.d	Preservice Inspection & Testing of Snubbers (MEB-21)	CLOSED	G02-81-313 9/24/81
3.9.2.5	Dynamic Model Justification (MEB-22)	CLOSED	*
3.9.3.1.a	Loading Combinations and Stress Limits (MEB-23)	CLOSED	N.L.U.
3.9.3.1.b	Allowable Stresses in Bolting (MEB-24)	CLOSED	*
3.9.3.1.c	Respond to Q110.027 (MEB-25)	CLOSED	N.L.U.
3.9.3.1.d	Method for Combining Responses (MEB-26)	CLOSED	*
3.9.3.1.e	Fatigue Analysis (MEB-27)	CLOSED	*
3.9.3.1.f	Fatigue Evaluation (MEB-28)	CLOSED	DAR
3.9.3.1.g	Justify Using 1 OBE (MEB-29)	CLOSED	*
3.9.3.1.h	Validity of 1.5 Sm (MEB-30)	CLOSED	*
3.9.3.1.i	Buckling Limits (MEB-31)	CLOSED	*
3.9.3.1.j	Basis for 1.5 AISC & 1.67 AISC (MEB-32)	CLOSED	*
3.9.3.1.k	General Membrane Plus Bending Allowable Stress (MEB-33)	CLOSED	*
3.9.3.1.l	Emergency Condition Criteria (MEB-34)	CLOSED	N.L.U.
3.9.3.1.m	Recirc. Pump Hanger Loads (MEB-35)	CLOSED	*
3.9.3.1.n	Additional Info Required (MEB-36)	CLOSED	*
3.9.3.1.o	Clarify Equation (MEB-36)	CLOSED	*
3.9.3.1.p	Justification of AISC Usage (MEB-37)	CLOSED	*
3.9.3.1.q	Explain Stress Limits (MEB-38)	CLOSED	*
3.9.3.1.r	Revise Table 3.9-2(y) (MEB-39)	CLOSED	*
3.9.3.1.s	Emergency Loading Condition (MEB-40)	CLOSED	*
3.9.3.2	NRC Site Audit	OPEN	NRC

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
3.9.3.3	Pressure Relief Devices (Q 110.031, MEB-41)	CLOSED	*
3.9.3.4.a	RPV Support Buckling (Q 110.29, MEB-42)	CLOSED	*
3.9.3.4.b	IEB 79-02 Base Plate Anchor Bolts (MEB-43)	CLOSED	*
3.9.4.a	Explanation of 3.9.4.3 (MEB-44)	CLOSED	*
3.9.4.b	Clarification of Table 3.9-2(u) (MEB-45)	CLOSED	*
3.9.5.a	Table 3-9-13 Stress Limits (MEB-46)	CLOSED	*
3.9.5.b	Cracking of BWR Jet Pump Holdown Beam (MEB-47)	CLOSED	G02-80-279 12/4/80
3.9.5.c	NUREG 0619 (Q121.8, Gen. ltr. 81-11) (MEB-48)	CLOSED	G02-82-36 1/13/82
3.9.6	Inservice Testing Program	CLOSED	G02-81-322 10/1/81
3.9.6	Isolation Valve Leakage (MEB-49)	CLOSED	G02-82-15 1/8/82
3.9.6	Isolation Valve Leak Rate (CSB-6, 7, 8)	CLOSED	*
3.10	Hydrodynamic Loading Effect	CLOSED	(Equip. Qual.) (4)
3.10	NRC Site Audit	OPEN	NRC
4.4	MCPR Operating Limit Calculation by ODYN	CLOSED	*
4.4	Description of Loose Part Diagnostic Procedures	CLOSED	*
4.4	Description of Operator Training on LPMS	CLOSED	*
4.6	Safety Concerns Associated with Pipe Break	CLOSED	G02-82-37 1/13/82
4.6	Safety & Operability of CRD Hydraulic Systems; Q10.43	CLOSED	G02-81-533 12/18/81
4.6	Revise Section 4.6 to incorp. BWR Scram Discharge Safety Evaluation; Q 10.41	OPEN	(ATWS) (5)
5.2.2	Overpressurization Protection (RSB-1)	CLOSED	*
5.2.2	SRV Surveillance (RSB-2)	CLOSED	LRG RSB-28 (6)
5.3.1	10CFR50 Appendix G	CLOSED	G02-81-532 12/18/81

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
6.1.2.II	Unqualified Protective Coatings; Q281.009 (CEB-3)	OPEN	1/31/82
6.2.1.8.e	Condensation Oscillation Load Report (CSB-43)	CLOSED	G02-81-552 12/24/81
6.2.1.8.g	Quencher Air Clearing Loads (CSB-44, 45, 46, 47 af)	CLOSED	G02-82-35 1/ 13/82
6.2.1.8.g (4)	SRV/LOCA and Chugging (CSB-48)	CLOSED	"
6.2.1.8.g (5)	Pool Temperature Limit Report (CSB-41)	CLOSED	G02-81-524 12/15/81
6.2.1.8.g (5)	In-plant SRV Test (CSB-47g)	CLOSED	"
6.2.4.3	Purge Valve Debris Screen (CSB-1)	CLOSED	*
6.2.4.3	Recombines Scrubber Return Line Second Isolation Valve (CEB-4)	CLOSED	*
6.2.5	Containment Inerting	OPEN	01/31/82
6.2.5	Inert Atmosphere H_2, O_2, N_2 Concentrations (CEB-1)	OPEN	2/7/82
6.2.6	Stear Valve Performance Assurance (CSB-21,22)	CLOSED	*
6.3	Pressure Interlocks on ECC Injection Valves (RSB-3, ICSB-9)	CLOSED	*
6.3	Premature LPCI Diversion (RSB-4)	CLOSED	*
6.3	Long Term Air Supply to ADS Valves (RSB-5)	CLOSED	*
6.3.2.3	Verify ECCS discharge line fill systems provided with continuous indication in control room	CLOSED	Q211.079 (7)
6.3.4	Verify RRC flow control valve control time	CLOSED	Q211.188 (7)
6.3.4	Verify RRC flow control valve opening post-LOCA	CLOSED	Q31.001(3), (7) Q31.058
6.3.4	Verify maximum peak cladding temperature increase	CLOSED	Q211.083 (7)

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
6.3.4	Verify recirculation flow control valve operation	CLOSED	RSB-4 (7)
6.5.1.2.1	R.G.1.52 Noncompliance	CLOSED	*
	Expand in Section 2.i of R.G.1.52	CLOSED	*
	Status of .3 Two-Inch Fiberglass Pads	CLOSED	*
	Cleanup Train Deviation	CLOSED	*
	Testing of Atm. Cleanup Components	CLOSED	*
	Testing Following Painting	CLOSED	*
	Flow Rate indication and High/Low Alarm	CLOSED	*
	Valve, Pressure Drop Indication	CLOSED	*
6.7	MSIV Leakage Rate	CLOSED	*
8.3.1	HPCS Diesel Reliability Testing (PSB-1)	CLOSED	*
8.3.1	Override of Test Mode (PSB-2)	CLOSED	G02-81-327 10/2/81
8.4.3	Thermal Overload Test Program (PSB-5)	OPEN	2/15
8.4.3	PSB BTP-1 (PSB-6)	OPEN	1/31
9.1.3	Fuel Pool Heat Loads Per BTP ASB 9-2	OPEN	1/31/82
9.1.4	Lifting of light objects	CLOSED	*
9.1.4	NUREG 0612	CLOSED	G02-82-32 1/13/82
9.4.1	CW System Writeup (ASB 9.4.1)	OPEN	
9.4.8	Tornado Missile Causing Loss of DG Area Cable Cooling (ASB 9.4.8)	CLOSED	*
9.4.12	Tornado Protection of Makeup Water Pump House Rents (ASB 9.4.12)	CLOSED	*
10.4.6	NUREG 0737/II B 3/Post Accident Sampling	CLOSED	II.B.3 (8)
11.4	Wet Solid Moisture Proportions of Cement	CLOSED	*
12.3.4	Access Monitors - Post Accident Radiation Monitors	CLOSED	*

SER SECTION	TOPIC (Branch Meeting Issue #)	STATUS	REFERENCE/ SCHEDULE
12.5.1	Education of HP Supervisor; Chem Tech	CLOSED	G02-82-25 1/11/82
12.5.1	Post Accident Monitor Range	CLOSED	*
13.5.2	Revise Chapter 13.5 to Reference R.G.1.33 Rev. 2	CLOSED	*
15	Thermal Power Monitor-in Transient Analysis (RSB-6)	CLOSED	*
15	ODYN Reanalysis (RSB-7)	CLOSED	G02-82-26 1/11/82
15	Transient Reclassification (RSB-8)	CLOSED	"
15.8	ATWS Emerg. Oper. Procedure	CLOSED	*
II.K.3.18	Modification of ADS Logic (RSB-9)/(ICSB-7)	CLOSED	II.K.3.18 (8)
II.K.3.21	Restart of Core Spray Systems (RSB-11)/(ICSB-8)	CLOSED	II.K.3.21 (8)
II.K.3.25	Loss of Power to Pump Seal Coolers (RSB-10)	CLOSED	II.K.3.25 (8)

BRANCH MEETING OPEN ITEMS

Containment Systems Branch

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE

Issues CSB-1, 6, 7, 8, 21, 22, 41, and 43-48 were addressed in the Draft SER.

CSB - 3, 5, 10, 13, 14, 17, 27, 28, 34, 35, 36, and 42 were closed at the 9/14/81-9/17/81 branch meeting. The following required further documentation to formally close out.

CSB-2	RFW Seal Water Acceptable Leak Rate	OPEN	02/15/82
CSB-4	Revise Figure 6.2-31a	CLOSED	*
CSB-9	Revise Table 6.2-16	CLOSED	*
CSB-11	Justify Who Not Two Isolation Valves On Recirc. Lines	CLOSED	*
CSB-12	Add check valve to Table 6.2-16	CLOSED	*
CSB-15	Revise Fig. 6.2-16	CLOSED	*
CSB-16	Debris screens for vacuum breakers	CLOSED	*
CSB-18	Correct Table 6.2-16	CLOSED	*
CSB-19,24	LPCI with Postulated LOCA w/o Alarm	CLOSED	*
CSB-20	Relocate Redundant Check Valves Inside Containment	CLOSED	*
CSB-23	Revise Fig. 6.2-31p	CLOSED	*
CSB-25,29	ILRT	CLOSED	*
CSB-26	Revise p. 6.2-50a and 6.2-107 (Q 022.071)	CLOSED	*
CSB-30	Revise Section 6.2.5.7 and 6.2.1.1.8.2	CLOSED	*
CSB-31	Revise p. 6.2-4	CLOSED	*
CSB-32	Test Airlock Doors	CLOSED	*
CSB-33	Subcooling Used in Containment Pressure Response	CLOSED	*
CSB-37	Subcompartment Pressurization	CLOSED	G02-82-03 1/6/82

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
CSB-38	R.G. 1.97	CLOSED	II.F.1
CSB-39	Include Load Definitions Section Information DAR Appendix	CLOSED	G02-82-34 01/13/82
CSB-40	Incorp. NUREG-0487 into Table C.1	CLOSED	*

Mechanical Engineering Branch

All issues except MEB-14 and 50 were addressed in the draft SER or are closed.

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
MEB-14	Provide Audit Materials for B&R Computer Program SRVDAM	OPEN	01/31/82
MEB-50	B&R sample certain systems when as-built drawings available	OPEN	6/01/82

Structural Engineering Branch

All issues except SEB-21 and Q 130.050 were addressed in the draft SER or were closed in our 12-14-81 submittal (G02-81-518).

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
SEB-21	Relative Displacement Between DG Oil Storage Tank & Connected Piping	OPEN	1/31/82
Q 130.050	Turbine Missile Study	CLOSED	G02-82-33 01/13/82

Chemical Engineering Branch

All issues were addressed in the draft SER.

I&C Systems Branch

The following issues resulted from the ICSB meeting of 12/10/81. (ICSB-9 was addressed in the draft SER under Section 6.3 and corresponds to RSB-3.)

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
ICSB-1	Redundant/Alternate Means of Remote Shutdown	OPEN	3/82
ICSB-2	Address RG.1.97 on Item by Item Basis in Section 7.5	CLOSED	G02-82-30 1/13/82
ICSB-3	Submit multiplexer article	CLOSED	*

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
ICSB-4	Reactor Level Instrument Line failures	CLOSED	*
ICSB-5a	BOP Compliance with IEB 80-06	OPEN	1/31
ICSB-5b	Control System Failures Questions		
	Q031.135 (IEB 79-27)	OPEN	10/1/82
	Q031.137 (IEN 79-22)	OPEN	10/1/82
	Q031.138	OPEN	10/1/82
	Q031.136 (IEB 80-06)	OPEN	4/82
ICSB-6	Relief & Safety Valve Position Indication	CLOSED	II.D.3
ICSB-7	Automatic Actuation of ADS	CLOSED	II.K.3.18
ICSB-8	Automatic Restart	CLOSED	II.K.13,21
ICSB-10	Spray Pond & SSW Pump I&C Design Description	CLOSED	*
ICSB-11	Explain Tables 7.3-3,5,7,& 23 (Q031.116)	CLOSED	*
ICSB-12	MSIV Leakage Control System	CLOSED	*
ICSB-A1	Recirculation Pump Trip Q031.115/CRG RSB-22	CLOSED	Q031.115/LRG RSB-22
ICSB-A2	Describe RPSMG Set EPAs	OPEN	1/31
ICSB-A3	Describe SDV Level Sensing Systems	CLOSED	Q10.41
ICSB-A4	Describe Design Features in Section 7.2	OPEN	1/31
ICSB-A5	Confirm Design's Seismic Qualification	CLOSED	*
ICSB-A6	Compliance with R.G.1.47	OPEN	1/31
ICSB-A7	Reference or Revise Series 3.1 Questions	OPEN	1/31

Power Systems Branch (Electrical)

Issues PSB-1, 2, 5, and 6 were addressed in the draft SER.

The following issues resulted from the PSB meeting of 12/17/81 (Issues PSB-3 and 4 were closed at the meeting).

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
PSB-E-7	Indicate Nonclass IE Loads are Isolated From Class IE Supply	OPEN	1/31
PSB-E-misc	D G Loading Sequence	CLOSED	*
PSB-E-8.3	Revise Chapter 8.3	CLOSED	G02-82-31 1/13/82

Power Systems Branch (Mechanical)

The following issues are open after PSB meeting of 12/16/81.

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
Q040.018	Diesel Skid Mounted Components' Design Data	OPEN	1/31
Q040.019/ 054/056	Air Pockets Vent to Expansion Tank; Discuss Reservoir Tank	OPEN	1/31
Q040.022/ 086	Air Start System Air Dryers	OPEN	1/31
Q040.023	L.O. Checks in Engine Starting Procedures	CLOSED	** (9)
Q040.026/ 058/059	Fire in Adjacent DG Room; DG Oil Bath Filter Maintenance	CLOSED	**
Q040.032/ 061	Monthly Valve Cycling	CLOSED	**
Q040.047	R.G. 1.137	CLOSED	**
Q040.050	Tank Internal Coating	OPEN	1/31
Q040.080	Heavy Duty Gear Drive Installation	OPEN	1/31
Q040.081	Equivalent Training for New Personnel	CLOSED	**
Q040.083	Removal of I&C Components from Skid	OPEN	1/31
Q040.084	Non Block Related Piping Code Class	OPEN	1/31
Q040.085	Keep Engine Block Warm	OPEN	1/31
Q040.087/ 088/089	Why Leave Upper Part of Engine Dry	OPEN	1/31

Reactor Systems Branch

All RSB issues were addressed in the draft SER.

Licensee Qualification Branch

LQB issues have been addressed separately.

Core Performance Branch (Telecon 1/6/82)

The issues below may be SER open items based on telephone conversations with NRC staff. A response was provided separately.

ISSUE #	TOPIC	STATUS	REFERENCE/ SCHEDULE
CPB-1	Channel Box Deflections	CLOSED	LRG CPB-7
CPB-2	Waterside Corrosion	CLOSED	LRG CPB-2
CPB-3	Seismic LOCA Loads Analysis	OPEN	6/30/82
CPB-4	Cladding Ballooning	CLOSED	---
CPB-5	On-Line Fuel Failure Detection	CLOSED	---
CPB-6	Post-Irradiation Examination	CLOSED	---

DRAFT SER OPEN ITEMS

NOTES:

(1) * = Response provided herein.

(2) NLU = Further documentation will be provided in an appendix to the FSAR in the New Loads Update.

(3) DAR = Further documentation will be provided in Revision 3 to the Design Assessment Report.

(4) Further documentation was provided in the Equipment Qualification submittal.

(5) Section 4.6 will be revised concurrent with description of ATWS modifications.

(6) Further documentation was provided in the LRG submittal (Appendix I).

(7) Refer to the indicated question or issue for a response to this item.

(8) Further documentation was provided in the TMI submittal (Appendix B).

(9) ** = Will be submitted with complete PSB-Mechanical Engineering package by January 31, 1982.

DSER 2.3.3

The onsite meteorological data system conforms to the guidance in Regulatory Guide 1.23 (Rev. 0) and provided adequate data to represent the onsite meteorological conditions as required in 10 CFR Part 100.10. The onsite data provide an acceptable basis for making conservative estimates of atmospheric diffusion for design basis accident and routine releases from the plant.

To address the meteorological requirements for emergency preparedness planning outlines in 10 CFR Part 50.47 and Appendix E to 10 CFR Part 50 the applicant will upgrade the operational meteorological program.

2.3.4 Short-term (Design Basis Accident) Diffusion Estimates

The dispersion of short-term (less than 30 days) accidental atmospheric releases from buildings and vents has been evaluated by the staff using an approach which is similar to the guidance provided in Regulatory Guide 1.145. The data submitted by the applicant (see Section 2.3.3) and vertical diffusion parameters developed from atmospheric diffusion data from desert field tests (Yanskey, et. al., 1966) were used. A ground level release and a building wake factor of $1383 m^2$ were assumed. The staff's 0.3 percentile direction dependent X/Q values are presented in Table 2.3-1.

The applicant has made estimates of the dispersion of gaseous effluents released during accidents. They used the onsite data and diffusion coefficients determined experimentally for stable atmospheric conditions at the Hanford Reservation. Diffusion coefficients for other stability categories were calculated as functions of wind speed and downwind distance with stability dependent exponents. The applicant used the

Open SER Issue

2.33 Emergency Preparedness Plan Meteorological Requirements

See revised FSAR page 2.3-37 (attached).

In several of the monthly summary reports, the computer programs as applied to dummy data have been compiled as called for in the Quality Assurance Manual[23] for the purpose of documenting proper programming and proper computer performance.

These computer computations have been verified with hand calculations made with the dummy data. The computational programs for χ/Q were similarly tested.

2.3.3.2.4 Meteorological Monitoring Program During Plant Operation

Insert attached

~~The WNP-2 operational meteorological program will commence at fuel load. The system will be put in operation at least two months prior to fuel load to ensure reliable operation at fuel load. System measurements will include wind speed and direction and temperatures at 245 and 33', Δt between 245 and 33', and dewpoint at 33'. A quality assurance program will be utilized to ensure measurement accuracies within those recommended by Regulatory Guide 1.23. Those parameters which will be multiplexed to the control room include wind speed and direction from 245' and 33' and the Δt between 245' and 33'. The control room displays will consist of instantaneous analog (strip chart) valves of each of the multiplexed parameters.~~

2.3.3.3 Other Meteorological Measurement Programs Considered for the Data Comparisons

2.3.3.3.1 WNP-2 Temporary Tower

A temporary 23 foot onsite tower was used during the period April 1, 1972 through August 31, 1974 to obtain data input for WNP-2 environmental studies and to provide a comparative overlap with the initially measured permanent tower data.

The temporary tower was located in the vicinity of the permanent towers with its base at approximately 448 feet MSL. Wind data from the temporary tower were obtained at the 23 foot level while temperature data were acquired at the three foot level. Wet bulb data from the temporary tower were established from techniques and data contained in the U.S. Department of Commerce, Weather Bureau Office Document: Relative Humidity and Dewpoint Table. As a special quality assurance program was not initiated for the temporary tower installation, it is not possible to assert that this tower's data complied with the requirements contained in Regulatory Guide 1.23.

Insert to Page 2.3-37:

The Supply System Meteorological Measurements Program (MMP) treats WNP-2 and WNP-1/4 as an integrated whole. As WNP-2 fuel load is scheduled prior to WNP-1/4 fuel load, the WNP-1/4 MMP will incorporate the procedures applied to WNP-2. The instruments and data acquired are described in Subsection 2.3.3.1. This data forms the primary input data which will be relayed to the control room and site computers, as required, on a real-time basis. Data is also multiplexed to the WNP-1/4 control room, the FFTF, and the PSP&L Skagit-Hanford site required by contract. The data is available to indicated locations as 5-minute average analog values and are converted to digital values for CRT and ~~teletype~~ printout displays with various analog meter displays also available. These digitized, electronically averaged, five-minute data will be processed into 15-minute averages for utilization in Supply System dispersion models. Longer period averages will also be computed for trend analysis and report generation. These data will be routed to satisfy display and processing requirements of the on-site Technical Support Centers and the Emergency Operations Facility (EOF). The primary meteorological tower data will be stored on tape at the tower and also stored for 24 hours in raw and processed form by the plant Data Acquisition System. Instrument calibrations and maintenance procedures will be implemented to meet the data recovery and system accuracy requirements of Regulatory Guide 1.23. The backup system will be sited near the EOF after appropriate consideration of local topography and the final EOF building configuration. Instrumentation, maintenance, calibration, and processing of sensor data will be identical to that of the primary system. Sensor data obtained from the backup system will be 10-meter, 5-minute averaged windspeed, wind direction, Sigma Theta, Temperature, dewpoint, a temperature difference (over a to-be-determined height) and a 5-minute precipitation total. This system is designed to meet or exceed a data unavailability of 0.01 and backup system data access of less than 5-minutes. All sensor data will be interactively accessable as detailed in the WNP-2 Emergency Preparedness Plan. Spatial data acquisitions are planned from a network of approximately twenty 10-meter towers operated by Battelle Pacific Northwest Laboratories for DOE. Data will also be obtained as available from the 400-meter HMS tower and auxillary systems operated by Battelle. Terrain data requirements per Regulatory Guide 1.23 will be satisfied by selecting the appropriate 10-meter tower data following written and automated selection procedures based on the current meteorological situation. Where economically feasible, the Supply System will acquire any additional data required for the safety of the public. Data exchange with state and federal agencies will be incorporated in the MMP to meet their requirements. The accuracy, calibration, and reliability of all data not directly controllable by the Supply System will be determined by the private/governmental controlling agency.

2.5.4.3 Evaluation of Foundations

Beneath all seismic Category I structural foundations, the existing upper loose sandy material was excavated down to the underlying very dense Ringold gravel and replaced in a denser state by compaction. The excavations to the Ringold formation extended down to 385 feet (msl) to 392 feet (msl) with some localized areas as deep as 375.8 feet (msl). The thickness of the compacted backfill and the main foundation features of the principal seismic Category I plant structures are shown on Table 1.

Compacted Backfill

The applicant informed the NRC Staff on April 22, 1981 and September 1, 1981 that there is a potentially reportable condition concerning soil backfilling, compaction and testing. The applicant's interim reports describing the deficiency indicate that the laboratory maximum density testing for the Class 1 backfill performed since May 1976 may have been performed incorrectly. The applicant is formulating a list of all the affected areas and will evaluate if any of these areas have to be retested. The staff will review the applicant's conclusions and provide evaluation of the applicant's study in a supplement to this SER.

Bearing Capacity

The applicant has provided for a safety factor in excess of 3 in calculating allowable static design bearing capacity. The staff agrees that this margin of safety is adequate for the support of the plant facilities.

Liquefaction Potential

The studies made by the applicant to evaluate liquefaction potential show that the foundation soils are not potentially liquefiable. The undisturbed Ringold gravel is very dense. Except for the areas under investigation, the compacted backfill has been placed to a relative density in excess of 75 percent and the backfill has been shown to be stable when subjected to the design safe shutdown earthquake loading of 0.25g effective peak acceleration (see Section 2.5.2 of the SER).

2.5.4.4 Conclusion

Based on the applicant's design criteria and construction reports and on the results of applicant's investigations, laboratory and field tests, and analyses presented in the FSAR, the staff has concluded that the site and plant foundations will be adequate to safely support the WPPSS Nuclear Project No. 2 (WNP-2), and to permit the safe operation of the ultimate heat sink system in accordance with the requirements of Appendix A to 10 CFR Part 100, pending satisfactory resolution of the open item identified in Section 2.5.4.5.

2.5.4.5 Open Item

The applicant informed the NRC staff in interim reports dated April 22, 1981 and September 1, 1981 that there is a potentially reportable condition concerning soil backfilling, compaction and testing. The report describing the deficiency indicates that the laboratory maximum density testing for the Class 1 backfill performed since May 1976 may have been performed incorrectly. The applicant is compiling a list of all the affected areas and plans to evaluate if any area needs retesting. The staff will review the applicant's final report and provide an evaluation in a supplement to this SER.

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

Docket No. 50-397

December 15, 1981
G02-81-527

Mr. R. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: NUCLEAR PROJECT 2
QUALITY CLASS I SOIL BACKFILL TEST PROGRAM

Reference: G02-81-462, November 12, 1981, GD Bouchey (Supply System) to RA Schwencer (NRC)

The Reference letter provided a summary of the efforts being taken to resolve a suspected deficiency concerning soil backfill placed since May 1976 for support of Quality Class I utilities and structures (remote air intake structures, remote air intake piping, standby service water piping, and electrical duct banks for the standby service water system). To resolve the concerns regarding this backfill, an extensive test program was conducted to determine in-situ properties of the soil. The attached report, "Evaluation of Quality Class I Utility Backfill, GA File 81-605", was prepared by Geologic Associates, Inc. and Burns and Roe to summarize the results of the test program and conclusions regarding the acceptability of the soil. The report concludes that although the existing soil does not meet specification requirements for relative density, the effect on safety-related utilities and structures is acceptable.

A copy of this report was given to Mr. Dinesh Gupta of the NRC, Geosciences Branch, during his visit to the WNP-2 site on December 7, 1981, to discuss this matter.

Based on the conclusions of the report, we have informed NRC, Inspection and Enforcement Branch, Region V, that this condition, which had been identified as a potentially reportable condition under the provision of 10CFR50.55(e), is not reportable.

With transmittal of this report, and subject to your review, the open item identified in the draft SER for WNP-2 is considered closed.

During a meeting with the NRC staff on this matter on October 7, 1981, the Supply System was requested to assess the significance of the soil backfill deficiencies in light of IE Circular No. 81-08, and advise the staff of our conclusions. IE Circular No. 81-08 is primarily concerned with insufficient compaction of foundation and backfill materials leading to excessive settlement of plant structures. As stated in a separate report by Shannon and Wilson dated May 11, 1976 (Referenced in FSAR Chapter 2.5), the compaction and backfill work performed for the major plant structures prior to May 1976 is acceptable. This is further verified by the excellent results of the settlement monitoring program in progress (Reference NRC question 362.10 and the response). Accordingly, it is concluded that the primary concern of IE Circular 81-08 (excessive settlement of major plant structures) is not a problem at the WNP-2 facility.

GD Bouchey (650)
Deputy Director
Safety and Security

EAF:kjf

Attachment: "Evaluation of Quality Class I Utility Backfill, WNP-2, Hanford, Washington", GA File 81-605 (3 copies)

cc: JA Forrest - B&R RO
RE Snaith - B&R NY
JJ Verderber - B&R NY
AI Cygelman - B&R 954W
FA MacLean - General Electric
S. Smith - General Electric
ND Lewis - EFSEC, Olympia
WS Chin - BPA
NS Reynolds - Debevoise & Liberman
OK Earle - B&R RO
E. Beckett - Nuclear Projects, Inc.
AD Toth - Resident Inspector
JA Satir - B&R NY
WNP-2 Files

Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. The missiles generated by natural phenomena of concern are those resulting from tornadoes. The applicant has identified a spectrum of missiles for a tornado zone III site as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The spectrum includes the weight and velocity of the postulated missiles and is in accordance with current tornado missile criteria. We have reviewed the missile spectrum and conclude that it is representative of missiles at the site and is, therefore, acceptable. Discussion of the protection afforded safety-related equipment from the identified tornado missiles is provided in Section 3.5.2 of this SER. Discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER. Based on our review of the tornado missile spectrum, we conclude that it was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guide 1.76 with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

General Design Criterion 4, "Environmental and Missile Design Bases," states that all structures, systems and components essential to the safety of the plant must be protected from the effects of externally generated missiles. Discussion of the tornado missile spectrum is contained in Section 3.5.1.4

of this SER. [The applicant has not committed to provide tornado missile protection from the utility pole to an elevation of 30 feet above the highest grade elevation within 0.5 miles of the facility structures as per the guidelines of Standard Review Plan 3.5.1.4. We cannot concur with the applicant's evaluation until this commitment is made.] All stored fuel is located within tornado missile protected reactor building and the spent fuel pool walls. This assures compliance with the specific guidance of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," relating to external missile protection for stored fuel. The piping to the ultimate heat sink is buried 7 feet below grade and backfilled with high density Class I fill to provide missile protection and, therefore, to conform to Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," relating to external missile protection for the ultimate heat sink. [Until the applicant provides an acceptable response to our question concerning SRP 3.5.14, we will be unable to confirm compliance with Regulatory Guide 1.117, "Tornado Design Classification," relating to specific protection of safety-related systems and components from tornado missiles.]

Based on the above, we conclude that the design of the plant for protection against externally generated missiles is in conformance with the guidelines of Regulatory Guides 1.13 and 1.27 concerning protection of the spent fuel and the ultimate heat sink from tornado generated missiles, and is, therefore acceptable. [Until the applicant verifies conformance with the Standard Review Plan 3.5.1.4 as it relates to protection of structures, systems and components from tornado generated missiles, we cannot conclude conformance with the requirements of General Design Criterion 4 with respect

SER Open Item 3.5.2
Tornado Missile Protection

General Design Criterion 4, "Environmental and Missile Design Basis," states that all structures, systems, and components essential to the safety of the plant must be protected from the effects of externally generated missiles. . . . The applicant has not committed to provide tornado missile protection from the utility pole to an elevation of 30 feet above the highest grade elevation within 0.5 miles of the facility structures as per the guidelines of Standard Review Plan 3.5.1.4. We cannot concur with the applicant's evaluation until this commitment is made.

Response:

The WNP-2 safety-related structures are designed to withstand the impact from the utility pole to an elevation of 30 feet above the highest grade within 0.5 miles of the site. A review of those safety-related structures exposed to these missiles results in the following findings:

a. A review of the topography indicates the highest finish grade within 0.5 mile of the plant is 460 ft.; therefore, the maximum elevation a utility pole can be driven to is elevation 490 feet.

b. Portions of the safety-related buildings exposed to these missiles have a minimum of 18 inches of concrete as compared to the requirement of the SRP (Region III) of less than 6 inches.

c. The velocities required for Region III are much lower than those considered in the WNP-2 design by a factor of over 2.0.

In view of the above conservative criteria for WNP-2, the integrity of the plant safety-related structures is ensured. The FSAR Section 3.5.1.4 is revised per the attached page to reflect the above.

Missile Description	Weight, LBS	Horizontal Impact Vel. Ft/Sec
a. Utility pole, 14 in. dia. butt x 35 ft. long	1600	241
b. Steel rod, 1 in. dia. x 3 ft. long	8	259

The design basis tornado generated missiles are considered to strike surfaces of structures in any direction. The 1600 lb. utility pole is considered to strike surfaces at any level up to a maximum level of 30 feet above ~~plant finish grade~~. The highest finish grade within 0.5 miles of the plant. The steel rod is considered to strike surfaces at any level above plant finish grade.

Figures 1.2-1 through 1.2-14 indicate the location of structures, equipment and components protected against tornado generated missiles.

3.5.1.4.1 Tornado Generated External Missiles

Structures which house systems, equipment and components essential to safe shutdown are designed to withstand the effects of design basis tornado generated missiles described in 3.5.1.4. These structures provide protection by the following means:

a. Reactor Building

The location of the reactor building with respect to the other plant structures is illustrated in Figures 1.2-3 through 1.2-7. Portions of the reactor building exterior walls are protected by adjacent structures against direct impact of tornado generated missiles, as indicated in Figure. 3.5-36. The exterior walls of the reactor building, up to the refueling floor at elevation 606'-10-1/2", are capable of withstanding the impact of the design basis tornado generated missiles. The exterior walls are constructed of 4 feet thick reinforced concrete to elevation 471'-0" which is 30 feet above plant finish grade.

D SER 3.6.2.a (MEB-1).

WNP-2 NUCLEAR PROJECT NO. 2

DRAFT SER

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

The review performed under this section pertains to the applicant's program for protecting safety-related components and structures against the effects of postulated pipe breaks both inside and outside containment. The effect that breaks or cracks in high and moderate energy fluid systems would have on adjacent safety-related components or structures are required to be analyzed with respect to jet impingement, pipe whip, and environmental effects. Several means are normally used to assure the protection of these safety-related items. They include physical separation, enclosure within suitably designed structures, the use of pipe whip restraints, and the use of equipment shields.

Our review under Standard Review Plan Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping", was concerned with the locations chosen by the applicant for postulating piping failures. We also reviewed for the size and orientation of these postulated failures and how the applicant calculated the resultant pipe whip and jet impingement loads which might affect nearby safety-related components.

The following discusses several open issues in our review and concludes with our findings which are contingent upon resolution of these open issues.

a. In order for us to complete our review, the applicant should provide a summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. This data is required for review to ensure that the pipe break criteria have been properly implemented. This data has not been submitted. Figures 3.6-11 through 3.6-36 are not completed. Therefore, review of these areas remains an open area.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 1

In order for us to complete our review, the applicant should provide a summary of the data developed to select postulated break locations including, for each point, the calculated stress intensity, the calculated cumulative usage factor, and the calculated primary plus secondary stress range. This data is required for review to ensure that the pipe break criteria have been properly implemented. Figures 3.6-11 through 3.6-36 are not completed.

Response:

Postulated pipe break locations were selected on a basis of significant change in flexibility in high energy piping systems. Examples of change in flexibility are pipe fittings (elbows, tees and reducers) and circumferential connections to valves and flanges. This method of selection was chosen since it was conservative and most expedient, not requiring the availability of detailed stress analysis of the piping systems. The use of this criteria could only result in the need for too many pipe whip restraints rather than too few. In cases where the design and installation of appropriate pipe whip restraints might prove to be difficult, an option always remained to evaluate the need for the restraint on the basis of stress criteria when the final stress analysis became available. The significance of the use of flexibility criteria for postulated pipe break locations is that it permitted the design of pipe whip restraints (if required) at an early date in the project.

Following selection of postulated break points, it was necessary to determine the movement of the pipe due to jet reaction. It was not always necessary to provide a pipe whip restraint for every postulated break. Where it was determined that the movement of the pipe did not strike any piping system and/or equipment necessary for safe shutdown of the reactor or necessary to mitigate the consequences of a LOCA and where the whipping pipe would not directly strike the primary containment vessel, then pipe whip restraint was not required. Where pipe whip restraint was required, the pipe whip restraint was designed to meet the unique conditions for the postulated break.

As indicated above, stress criteria referred to in the question did not enter into the establishment of the basic design for WNP-2. However, additional studies are now underway wherein stress criteria for determination of break locations are being applied. Stress-related information used in connection with these studies will be provided when it becomes available upon the completion of these studies.

Summation - In connection with the additional studies discussed in paragraph 3 of the response, the results of these studies will be supplied in the FSAR together with specific stress criteria to complete Figures 3.6-11 through 3.6-36.

DSER 3.6.2.b (MEB-2)

b. Paragraph 3.6.2.1.1.1.b (2)(b)(Page 3.6-25) implies that the cumulative usage factor limit of 0.1 is considered only when high stress occurs. It is the staff's position that breaks must be postulated at any location where the cumulative usage factor exceeds 0.1. At these locations both circumferential and longitudinal pipe breaks should be postulated, unless it can be clearly shown that the high usage factor is due primarily to stresses in only one principle direction. The applicant response to Question 110.012 states that the rules set forth in 3.6.2.1.4.1.e (1) and (2) exempt certain break orientations based solely on stress and are independent of calculated cumulative usage factor. Clarification of this area is required.

c. Paragraph 3.6.2.1.1.1.b (2)(c)(Page 3.6-25) implies that breaks are postulated when the stress ranges as calculated by Equations 12 or 13 of the code exceed 2.4 S_m and Equation 10 exceeds 3 S_m. It is the staff's position that if Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

Eq. (10) > 2.4 S_m and Eq. (12) > 2.4 S_m

or

Eq. (10) > 2.4 S_m and Eq. (13) > 2.4 S_m

d. For those portions of ASME, Section III, Class 1 piping discussed in FSAR Sections 3.6.2.12.1 and designed to seismic Category I standards and included in the break exclusion area breaks need not be postulated providing all of the following criteria are met.

(1) Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not exceed 2.4 S_m.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 2

It is the staff's position that breaks must be postulated at any location where the cumulative usage factor exceeds 0.1. At these locations, both circumferential and longitudinal pipe breaks should be postulated, unless it can be clearly shown that the high usage factor is due primarily to stresses in only one principle direction. The applicant's response to Q. 110.012 states that the rules set forth in 3.6.2.1.4.1e (1) and (2) exempt certain break orientations based solely on stress and are independent of calculated cumulative usage factor. Clarification of this area is required.

RESPONSE

Where cumulative usage factor is a determinant in establishing a postulated break location, then to determine whether both a circumferential and longitudinal break need be postulated, the stresses in the two directions are compared.

FSAR page change (3.6-30a) agreed.

"(1) If the result of a detailed stress analysis indicates that the maximum stress range in the axial direction is at least 1.5 times that in the circumferential direction, only a circumferential break is postulated. Where usage factor is a determinant in establishing a postulated break location, the fatigue dominant stresses are examined as indicated above to determine whether longitudinal, circumferential or both are postulated"

Summation - This item is closed.

See revised FSAR page 3.6-30a (attached).

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

Where usage factor is a determinant in establishing a postulated break location, the fatigue dominant stresses are examined as indicated above to determine whether longitudinal, circumferential, or both are postulated.

D SER 3.6.2.c (MEB-3)

b. Paragraph 3.6.2.1.1.1.b (2)(b)(Page 3.6-25) implies that the cumulative usage factor limit of 0.1 is considered only when high stress occurs. It is the staff's position that breaks must be postulated at any location where the cumulative usage factor exceeds 0.1. At these locations both circumferential and longitudinal pipe breaks should be postulated, unless it can be clearly shown that the high usage factor is due primarily to stresses in only one principle direction. The applicant response to Question 110.012 states that the rules set forth in 3.6.2.1.4.1.e (1) and (2) exempt certain break orientations based solely on stress and are independent of calculated cumulative usage factor. Clarification of this area is required.

c. Paragraph 3.6.2.1.1.1.b (2)(c)(Page 3.6-25) implies that breaks are postulated when the stress ranges as calculated by Equations 12 or 13 of the code exceed 2.4 S_m and Equation 10 exceeds 3 S_m. It is the staff's position that if Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

Eq. (10) > 2.4 S_m and Eq. (12) > 2.4 S_m

or

Eq. (10) > 2.4 S_m and Eq. (13) > 2.4 S_m

d. For those portions of ASME, Section III, Class 1 piping discussed in FSAR Sections 3.6.2.12.1 and designed to seismic Category I standards ~~and~~ included in the break exclusion area breaks need not be postulated providing all of the following criteria are met.

(1) Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not exceed 2.4 S_m.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 3

For ASME, Section III, Class 1 piping designed to Seismic Category I standards, breaks due to stress are to be postulated at the following locations:

a. If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 Sm, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 Sm, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$
or
$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

b. Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

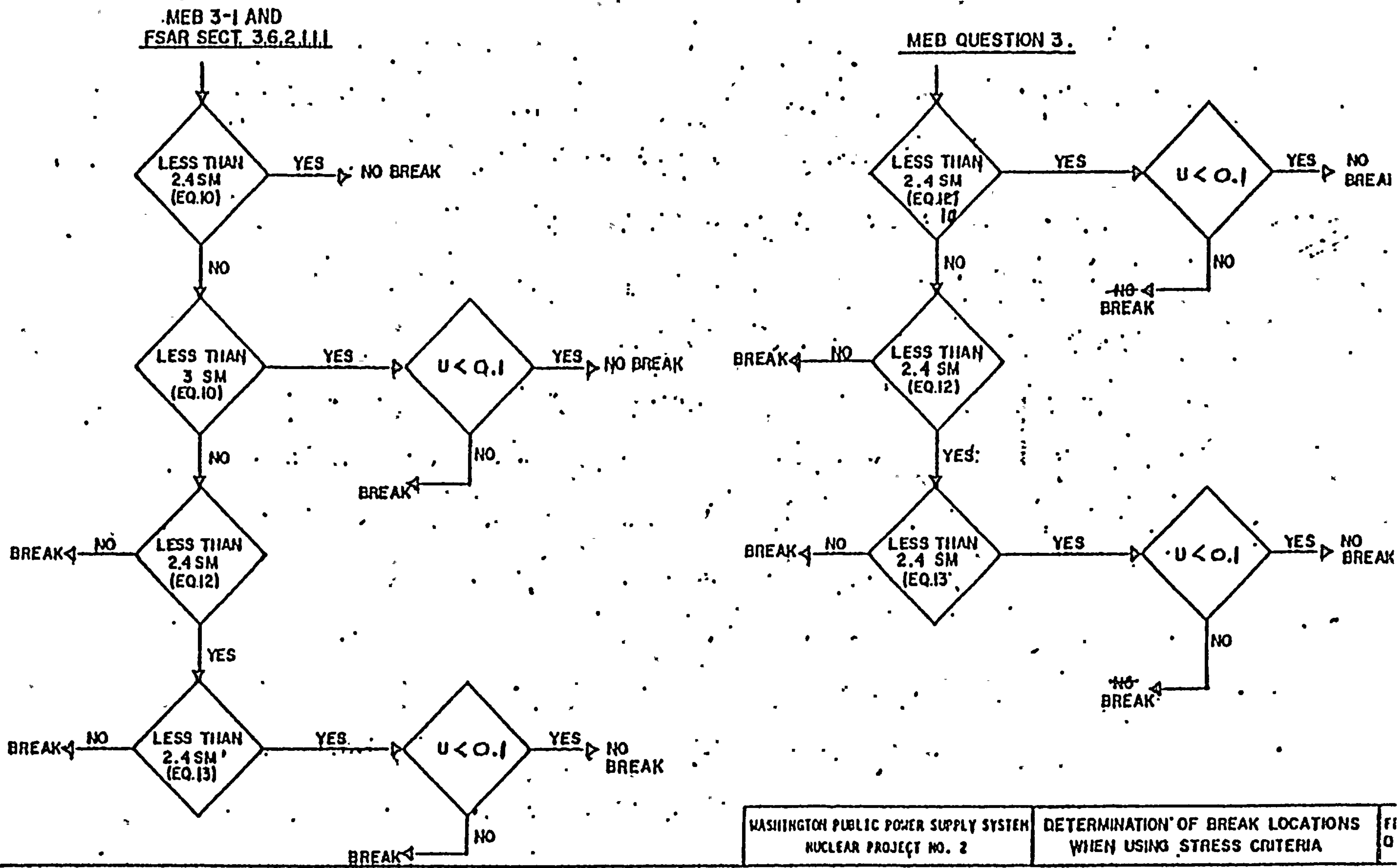
The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE. Any deviations from the above criteria must be justified.

Response:

WNP-2 will determine break locations for ASME, Section III, Class 1 piping consistent with the NRC position given in Branch Technical Position MEB 3-1.

Summation - This item is closed.

ASME SECTION III, SSI PIPING



Note: SM = Sm

DSER 3.6.2.d (MEB-4)

(2) If Eq. (10) does exceed 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if:

Eq. (10) > 2.4 S_m and Eq. (12) < 2.4 S_m

and

Eq. (13) < 2.4 S_m

(3) The cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, deadweight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration areas must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all ASME Class 1, 2 and 3 piping in the break exclusion area. It is not clear whether footnote (a) on Page 3.6-28 of the FSAR is applicable to Section 3.6.2.1.2.2.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

A list of all systems included in the break exclusion areas must be included in the FSAR. In addition, break exclusion areas should be shown on the appropriate piping drawings.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 4

A. For those portions of ASME Section III, Class 1 piping discussed in FSAR Section 3.6.2.1.2.1 and seismic Category I standards and included in the break exclusion area, breaks need not be postulated providing all of the following criteria are met:

a. Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III, does not exceed 2.4 Sm.

b. If Eq. (10) does exceed 2.4 Sm, then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 Sm, a break need not be postulated. In other words, a break need not be postulated if:

Eq. (10) $> 2.4\ S_m$ and Eq. (12) $< 2.4\ S_m$ and

Eq. (13) $< 2.4\ S_m$

c. The cumulative fatigue usage factor is less than 0.1.

d. For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3552 under the loadings of internal pressure, deadweight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including OBE.

In addition, augmented inservice inspection is required on all ASME Class 1, 2 and 3 piping in the break exclusion area. It is not clear whether footnote (a) on page 3.6-28 of the

FSAR is applicable to Section 3.6.2.1.2.2. The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.3. A list of all systems included in the break exclusion areas must be included in the FSAR. In addition, break exclusion areas should be shown on the appropriate piping drawings.

B. a. Document the method used to verify that the stresses in welded flued head fittings meets the limits specified in MEB 3-1. Indicate on which piping system the welded flued heads are used.

b. Describe the inservice inspection of the welded flued head fittings.

Response:

A. Footnote (a) on page 3.6-28 is not applicable to 3.6.2.1.2.2. The revision to page 3.6-28 indicates the systems with break exclusion areas between primary containment isolation valves. These systems are ASME Section III Class 1 systems.

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*With respect to loadings resulting from postulated piping failure outside the exclusion area (Section B.1.b.(1)(d) of MEB 3-1) detailed analyses were performed for the main steam and feedwater systems for breaks inside and outside of containment. These analyses have confirmed the acceptability of main steam and feedwater systems and have been used to conclude that detailed analyses for smaller lines are not required.

Please refer to revised FSAR page 3.6-28 for a list of all systems included in the break exclusion area. See Figures 3.6-147a through 3.6-147e for break exclusion areas.

B.a. A project unique analysis of the welded flued head fitting in the mainsteam penetration has been performed to assure meeting the code stress limits and the NRC criteria for the break exclusion area and documented in the WNP-2 Stress Report. In addition, GE has demonstrated the physical integrity of the WNP-2 flued head by a bounding generic finite element analysis on a similar configuration; this is documented in the GE Report NEDO-23652.

b. The flued head to process pipe weld is examined volumetrically from axial and perpendicular surfaces using ultrasonic methods. By using 45 and 60 degree shear wave from the perpendicular surface and 0 degree longitudinal wave from the axial surface complete coverage of the weld is assured. Details of the examination are contained in procedure UTP-33, which is contained in the WNP-2 PSI Program Plan. In addition, surface examination will be performed on the accessible portion of the weld.

Summation - This item is closed.

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

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

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

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

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

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

A Insert attached

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

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

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

3.6.2.1.2.3 Primary Containment Penetration Piping

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

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

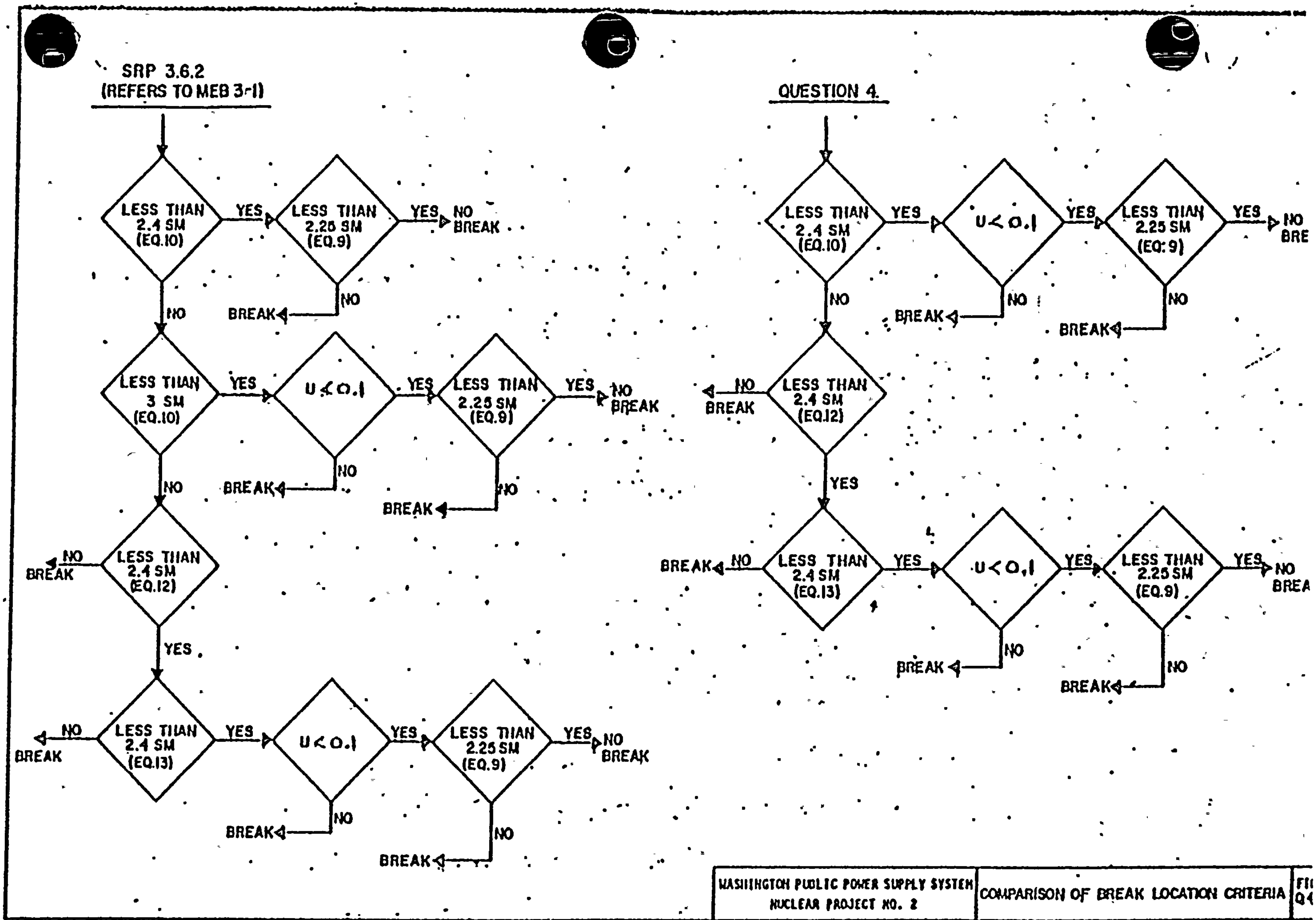
Insert to Page 3.6-28:

Piping systems which may have break exclusion areas between primary containment isolation valves are those determined by examining the list of high energy piping systems (see 3.6.2.1 and Table 3.6-2). Systems which do not pass through primary containment are excluded. In addition, systems which are not pressurized between the isolation valves during normal plant operation (see 3.6.2) are excluded. The remaining systems, those which may have break exclusion areas between primary containment isolation valves, are listed in Table 3.6-18. Break exclusion areas for these systems are shown on Figures 3.6-147a through 3.6-147e.

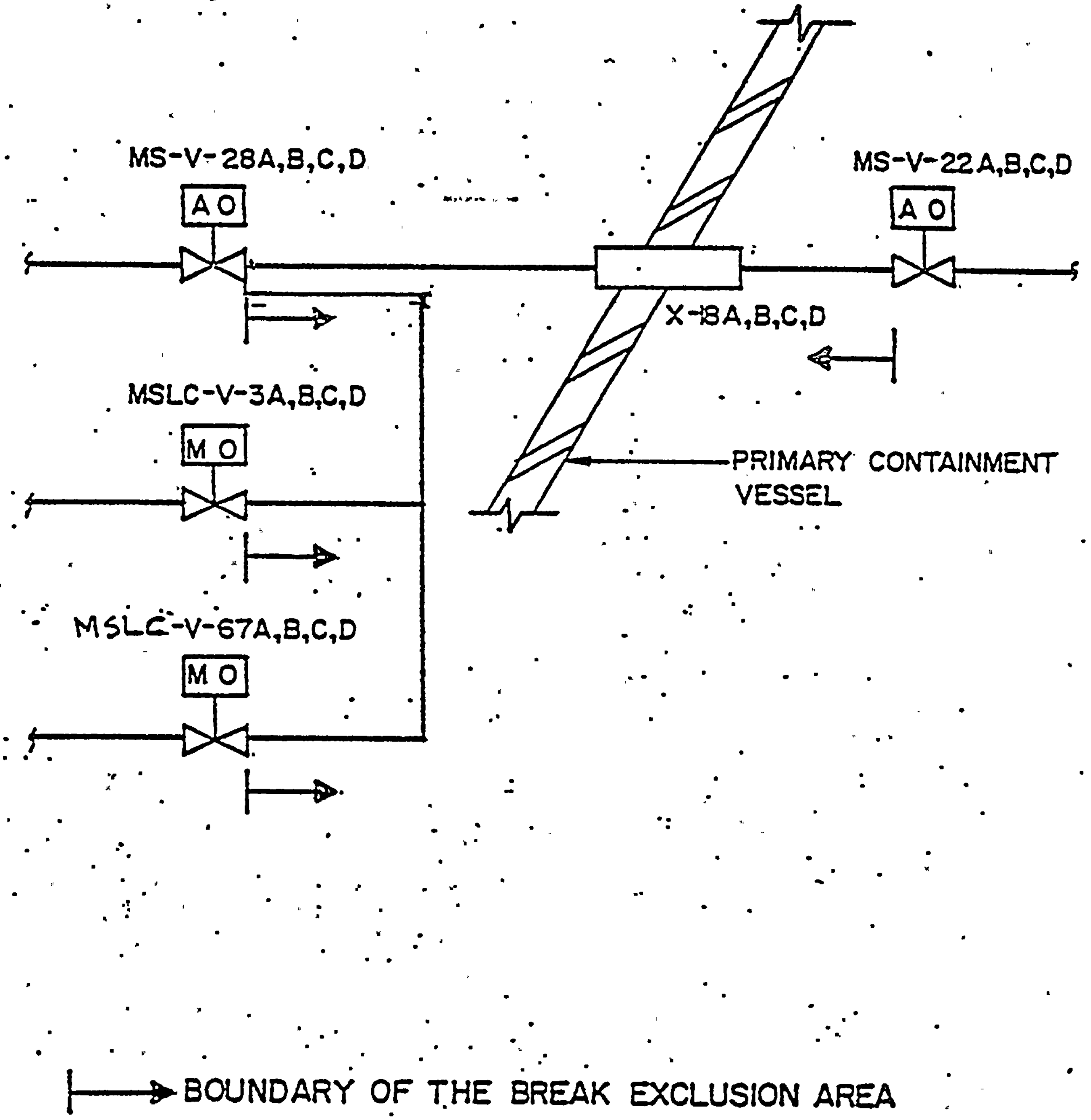
TABLE 3.6-18

PIPING SYSTEMS CONTAINING BREAK EXCLUSION
AREAS BETWEEN PRIMARY CONTAINMENT ISOLATION VALVES

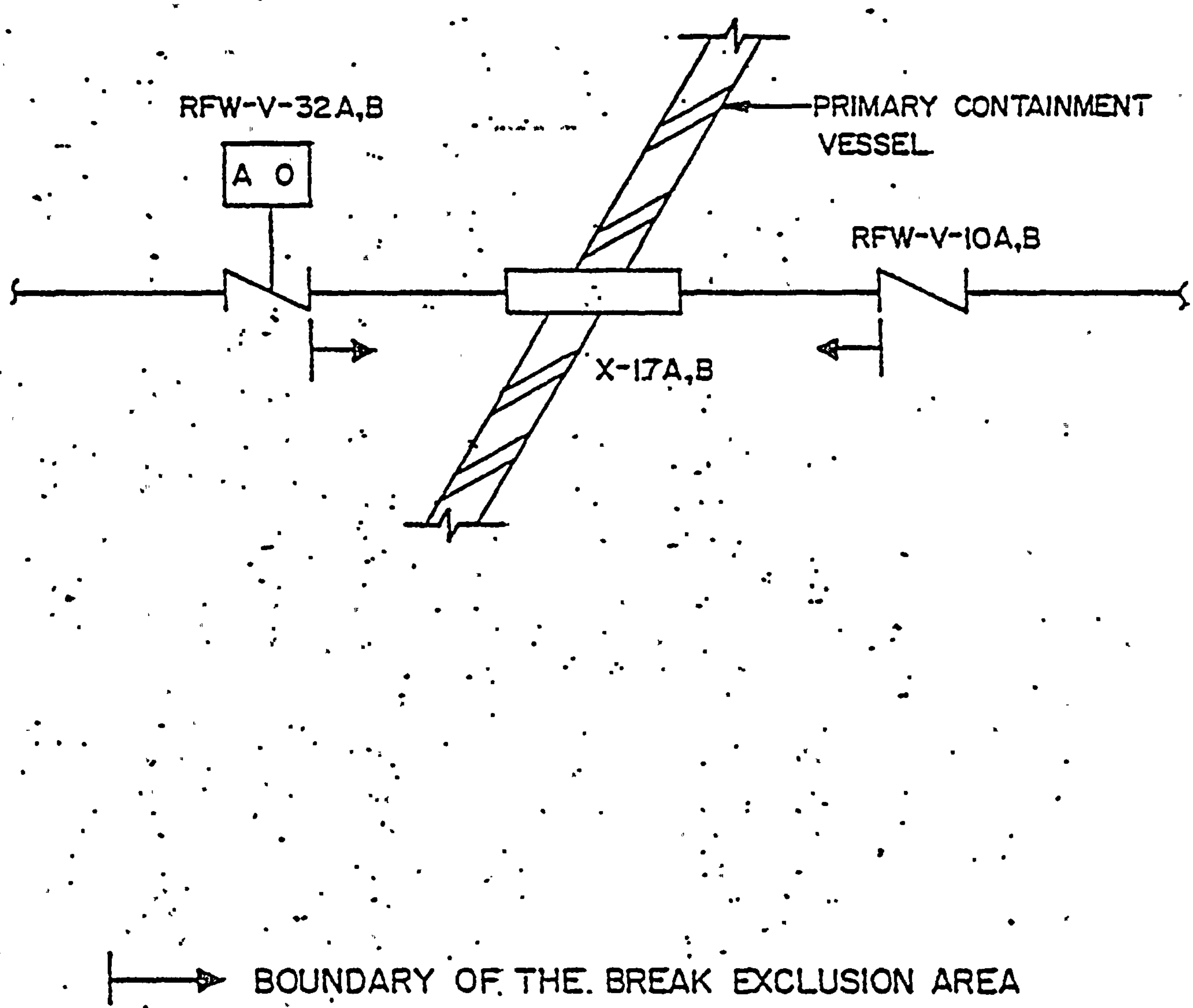
PIPING SYSTEM	PIPE SIZE
Main Steam Loop A	26"
Main Steam Loop B	26"
Main Steam Loop C	26"
Main Steam Loop D	26"
Reactor Feedwater Line A	24"
Reactor Feedwater Line B	24"
RHR Condensing Mode/ RCIC Turbine Steam	10"/4"
Reactor Water Cleanup	6"
~~Main Steam Valves Drainage Piping~~	~~3"~~


SRP 3.6.2
(REFERS TO MEB 3-1)
QUESTION 4.
LESS THAN 2.4 SM (EQ.10)
LESS THAN 2.25 SM (EQ.9)
LESS THAN 3 SM (EQ.10)
LESS THAN 2.4 SM (EQ.12)
LESS THAN 2.4 SM (EQ.13)
U<0.1
YES
NO
BREAK
NO BREAK
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2
COMPARISON OF BREAK LOCATION CRITERIA

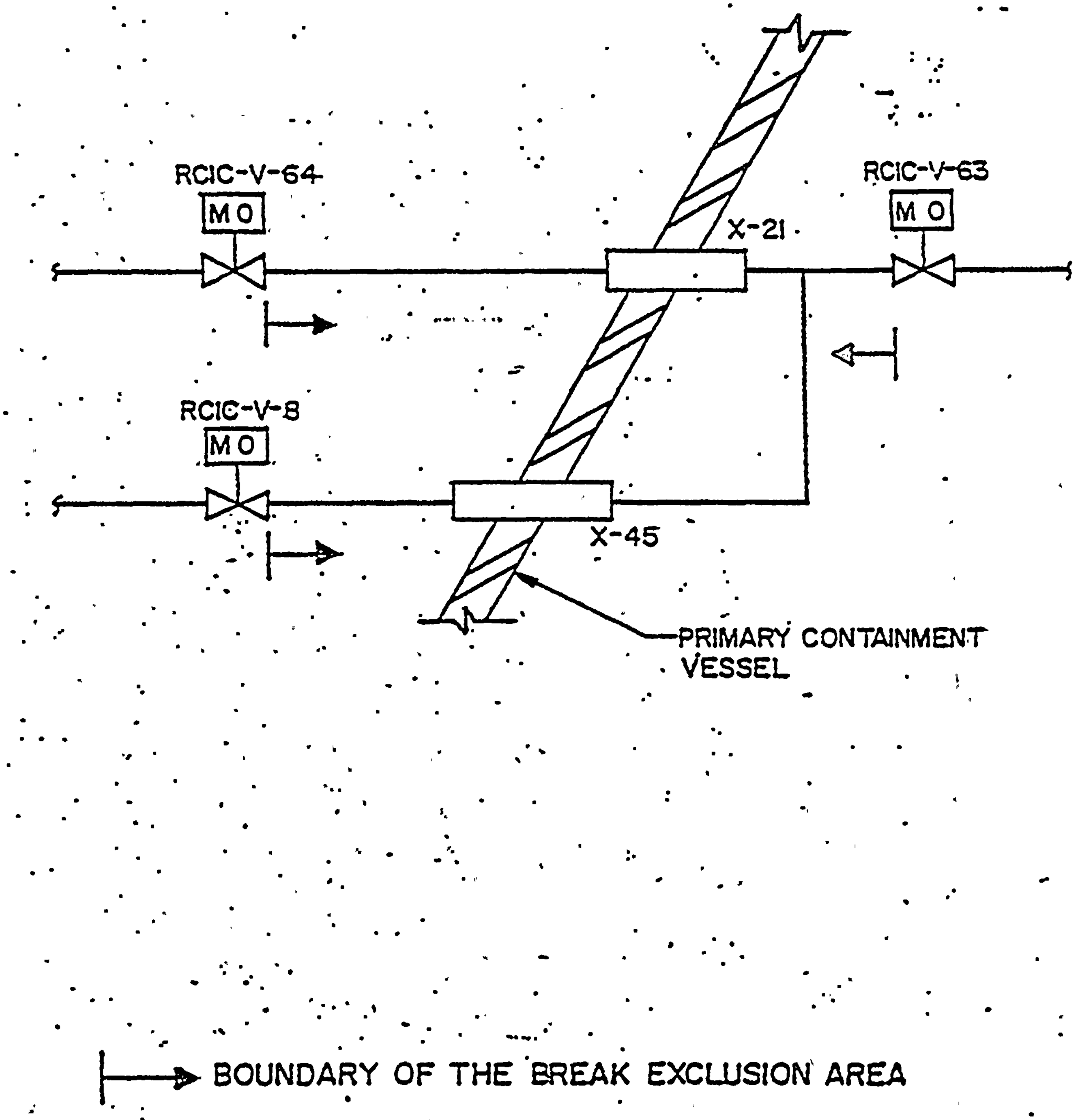
Note: SM = Sm



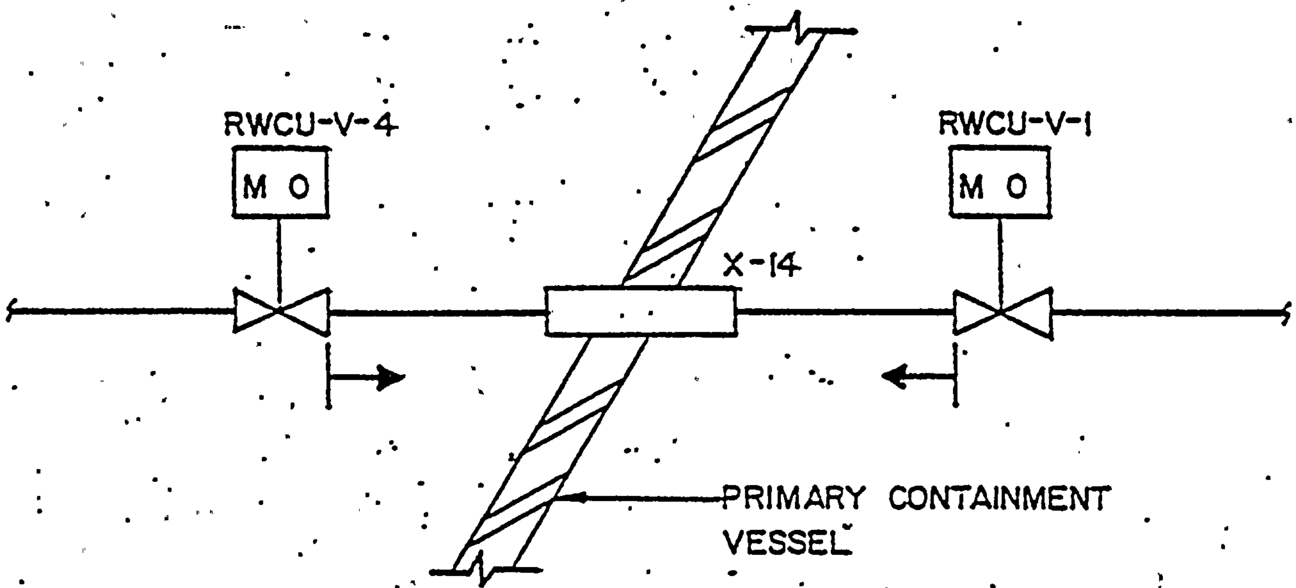
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM PIPING SYSTEM	FIGURE 3.6-147a



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR FEEDWATER PIPING SYSTEM	FIGURE 3.6-147b

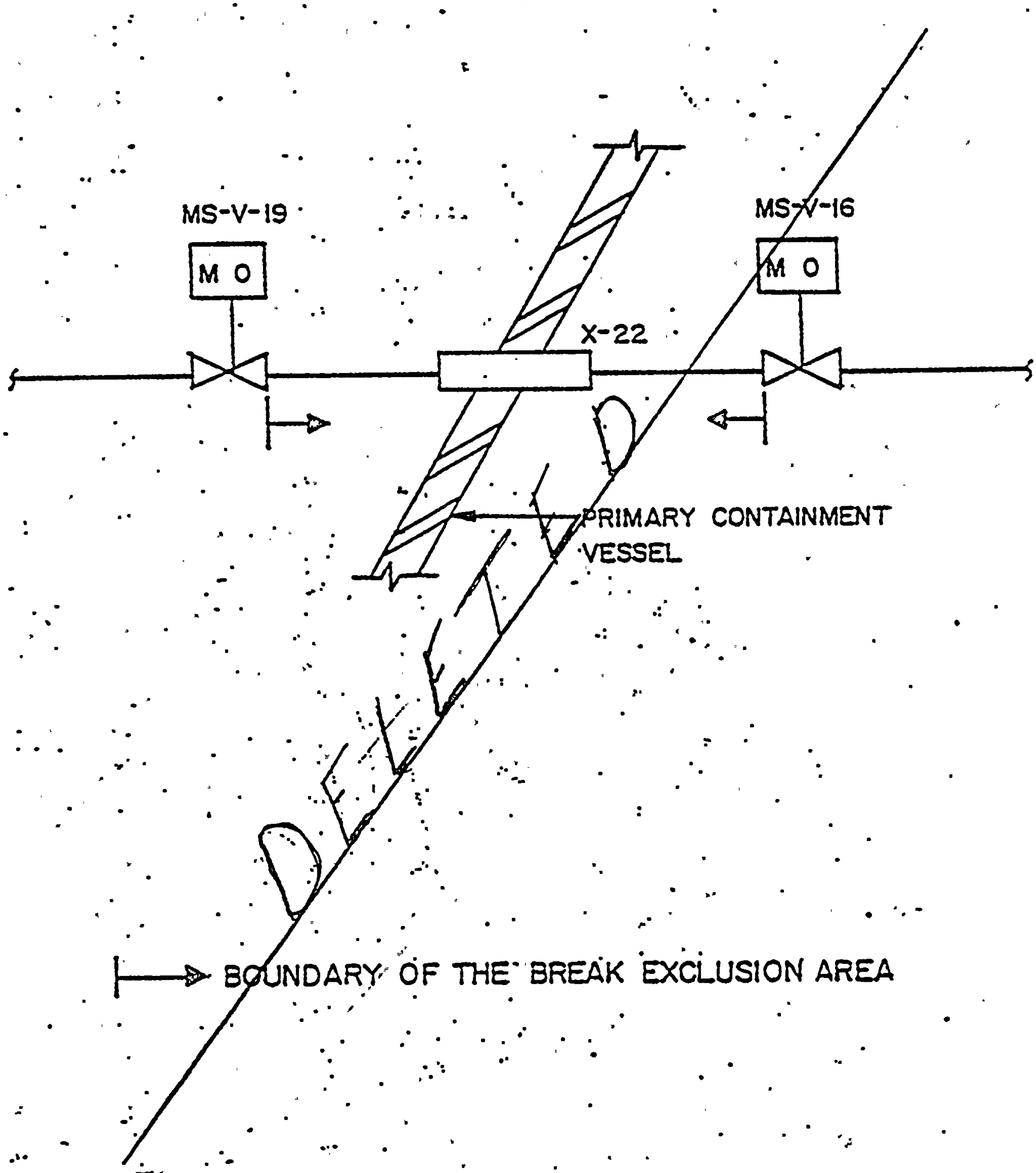


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR CONDENSING MODE/RCIC TURBINE STEAM PIPING SYSTEM	FIGURE 3.6-147c



BOUNDARY OF THE BREAK EXCLUSION AREA

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR WATER CLEAN UP PIPING SYSTEM	FIGURE 3.6-147d



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM VALVES DRAINAGE PIPING SYSTEM	FIGURE 3.6-147e

DSER 3.6.2.e (MEB-5)

e. Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

f. Paragraph 3.6.2.5.4.11 c (Page 3.6-70) states, "A pipe break in one of the six lines, if unrestrained, may result in pipe whip impact with adjacent isolation valves, possibly rendering them inoperative. Furthermore, unrestrained motion may cause impact with other lines, which may result in escalation of pipe breaks. Such a condition may unacceptably increase the severity of the initial pipe break." The way this paragraph is written, it is not apparent that sufficient protection has been provided to preclude the failure conditions discussed or whether these are failure conditions for which the protection was provided. Clarification of this area is requested.

Subject to resolution of the above open issues, our findings are as follows:

The applicant has proposed criteria for determining the location, type and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. The applicant has used the effects resulting from these postulated pipe failures to evaluate the design of systems, componenets, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. The applicant has stated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated piping failures.

We have reviewed these criteria and have concluded that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they will function

QUESTION NO. 5
(3.6.2)

Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

RESPONSE

In all analyses, except annulus pressurization analyses, full break with area equivalent to the pipe cross section is postulated to occur instantanesouly. No mechanistic approach is used.

In analyses related to annulus pressurization, the instantaneous approach is used in jet impingement and pipe whip restraint loads calculation. For the pressure time history, the recirculation line is postulated to break instantaneously producing full blowdown force. Subsequently, the broken end is assumed to separate in a finite time based on momentum and energy consideration. These analyses will be documented in detail as an appendix to the FSAR in the New Loads update.

Summation - This item is closed.

DSER 3.6.2.f (MEB-6)

e. Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

f. Paragraph 3.6.2.5.4.11 c (Page 3.6-70) states, "A pipe break in one of the six lines, if unrestrained, may result in pipe whip impact with adjacent isolation valves, possibly rendering them inoperative. Furthermore, unrestrained motion may cause impact with other lines, which may result in escalation of pipe breaks. Such a condition may unacceptably increase the severity of the initial pipe break." The way this paragraph is written, it is not apparent that sufficient protection has been provided to preclude the failure conditions discussed or whether these are failure conditions for which the protection was provided. Clarification of this area is requested.

Subject to resolution of the above open issues, our findings are as follows:

The applicant has proposed criteria for determining the location, type and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. The applicant has used the effects resulting from these postulated pipe failures to evaluate the design of systems, componenets, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. The applicant has stated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated piping failures.

We have reviewed these criteria and have concluded that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they will function

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Question 6.

Expand paragraph 3.6.2.5.4.11c to provide assurance that sufficient protection has been provided to preclude the pipe break damage for main steam and reactor feedwater piping inside the main steam tunnel.

Response:

Please refer to revised 3.6.2.5.4.11 for the information requested.

Summation - This item is closed.

c. Verification of Pipe Whip Protection Adequacy

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

3.6.2.5.4.11 Main Steam and Reactor Feedwater Piping Inside Main Steam Tunnel

a. System Arrangement

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

b. Pipe Whip Protection

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

c. Verification of Pipe Whip Protection Adequacy

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

Insert

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

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

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

a. System Arrangement

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

b. Pipe Whip Protection

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

c. Verification of Pipe Whip Protection Adequacy

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

Insert to Page 3.6-70:

Pipe whip restraints are provided to prevent pipe whip impact with the main steam or feedwater isolation valves. In addition, impact with adjacent main steam or feedwater lines is prevented. Refer to Figures 3.6-6g through 3.6-6k.

spaced frequencies, assessment was made by the applicant and shown to meet the requirements of Regulatory Guide 1.92.

The absolute sum (ABS) of two earthquake components of the maximum codirectional responses was used instead of SRSS of three components of the earthquake motion for both the time history and response spectrum methods. Comparisons of the results obtained from both the ABS and SRSS methods were made by the applicant and it was demonstrated that for all frequencies larger than 1.25 Hz the ABS method used by the applicant is conservative. This finding is acceptable to the staff since the frequency range of interest/concern in WNP-2 Category I structures and systems is always larger than 5 Hz.

The present technical position of the staff requires that the accidental torsion, minimum of 5% of the base dimension, be included in the design of structures. This is in addition to that which results from the actual geometry and mass distribution of the building. In response to staff request, the applicant provided calculation of design margin accounting for the accidental torsion for all Category I structures and showed that even for the structures with the lowest design margin, the factor safety values change by less than 2% and are still adequate. This is acceptable to the staff.

Floor spectra inputs used for design and test verifications of structures, systems, and components were generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis is employed for all structures, systems, and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning are considered.

The lumped mass-spring method was used to evaluate soil-structure interaction and structure-to-structure interaction effects upon seismic responses. However, the current staff position regarding the soil-structure interaction requires, in addition to the use of elastic half-space approach, the use of finite element method. The applicant provided the comparisons of the original soil spring analysis versus the finite element approach at different key locations in the reactor building and concluded that the soil spring analysis results envelop those from the finite element method. Furthermore, the applicant will perform

the analysis using the two different methods for radwaste building and submit the results by November 13, 1981 for staff review and acceptance.

The applicant used the equivalent static analysis for the spray ponds retaining wall and slabs and committed to provide analysis procedures and calculations demonstrating the conservatism of the method used for staff review and acceptance.

The acceptance of the applicant's seismic system and subsystem analysis is pending on the resolution of the above cited items.

3.7.4 Seismic Instrumentation Program

The type, number, location, and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude, and phase relationship of the seismic response of the Category I structures comply with Regulatory Guide 1.12. Supporting instrumentation is being installed on Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

The installation of the specified seismic instrumentation in the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic responses in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12.

Open SER Issue

3.7.3(b) Spray Pond Analysis; Retaining Wall Conservation (SEB-32)

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT.

WPPSS CORRESPONDENCE NO. ______

GD Bouchey - 370
BA Holmberg - 906D
RG Matlock - 901A
RM Nelson - 906D
GC Sorensen - 340
Waddel - 405
ket File
~~Chrono File~~
~~CD Taylor - 906D~~
~~GDB/LB - 370~~
~~BAH/LB - 906D~~
GCS/LB - 440
CDT/LB
sf(2)
pf
~~J Yatabe - 410~~
PL Powell - 906D
~~kf/file~~
RMN/LB

bcc: EF Beckett - NPI
OK Earle - B&R
J Plunkett - NUS
NS Reynolds - D&L
WNP-2 Files

Docket No. 50-397

December 14, 1981
G02-81-518
SS-L-02-CDT-81-109

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
OPEN ITEMS FROM SEB MEETING

Attached are sixty copies of the open items from the Structural Engineering Branch meeting held in Woodbury, New York, October 5 - 8, 1981.

Very truly yours,

Original Signed by:

G. D. Bouchey, Deputy Director
Safety and Security

CDT/rch
Attachments

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

FILE COPY

AUTHOR:	CD Taylor		FOR SIGNATURE OF:	GD Bouchey	
SECTION					
FOR APPROVAL OF	RM Nelson	BA Holmberg	GC Sorensen		
APPROVED	RM Nelson	BA Holmberg	GC Sorensen		
DATE			12/10/81		

DSER 3.7.3.a (MEB-7)

The applicant's procedures for the dynamic analysis of Category I systems, components, equipment and their supports have been reviewed by us and found to be generally acceptable. However, the following open issues must be resolved before we can report our findings.

a. Paragraph 3.7.2.1.8.2 (Page 3.7-15) stated that for the equivalent static load method, a minimum load factor of 1.15 is applied to building accelerations to include the effect of higher modes of vibration. The acceptance criteria of SRP 3.7.2 for the equivalent static load method is to apply a load factor of 1.5. A factor of less than 1.5 may be used if adequate justification is provided. Justification for utilizing this reduced factor is required.

b. Paragraph 3.7.3.2.1 of the FSAR states that "Based on Reference 3.7-10 (BWR/6 General Electric Standard Safety Analysis Report, Volume 1, General Electric Company, 4/30/74), which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event." Please provide clarification of this statement.

c. Paragraph 3.7.3.2.2 arrives at only one OBE intensity earthquake for design of the NSSS systems and components. Justification is required for this conclusion. Specifically, provide justification that the information in paragraph 3.7.3.2.2 is applicable to the WNP-2 site.

QUESTION NO. 7

Provide justification for utilizing the load factor of 1.15 for the equivalent static load method. The acceptance criteria of SRP 3.7.2 for the equivalent static load method is to apply a load factor of 1.5.

RESPONSE

a. Paragraph 3.7.2.1.8.2 has been revised to clarify the alternate simplified method of analyses. (See attached revision).

b. A summary from the study performed to verify the adequacy of the alternate simplified method is attached.

Summation - This item is closed.

An alternate simplified method of dynamic analysis is used for cold and/or limber piping systems. This is the Equivalent Static Load Method for piping. This method consists of applying constant horizontal and vertical load factors conservatively derived from seismic floor response spectra.

The description of the method is as follows: Enveloped seismic building response spectra are derived from widened seismic floor response spectra. (The widening of the building response spectra is described in 3.7.2.5).

The piping is supported seismically such that the ~~piping fundamental frequency is higher than the building fundamental frequency, i.e., the piping system is more rigid than the building.~~ first fundamental mode of all piping systems is above the spectral peak of the seismic response spectrum.

The piping systems are then represented by simply analytical models, e.g., simply supported beams. Initial maximum seismic support spans are analytically determined from the above model for the piping fundamental frequency. These maximum spans are modified, if required, so as not to exceed a conservative value of maximum stress based on ASME Code allowables, and a limiting piping deflection between supports.

insert attached

~~The building accelerations at the piping frequency are determined from the enveloped seismic building response spectra. These accelerations are increased by a minimum factor of 1.15 to include the effect of higher modes of vibration. These increased accelerations (g-values) are the load factors.~~

~~The horizontal and vertical loading factors are combined in the same way as described above for the detailed dynamic analysis.~~

3.7.2.1.8.3 Dynamic Analysis of Equipment

Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Category I equipment are given in Table 3.9-2. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response specrum analysis is performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance of plant using the envelope of response spectra for supports. Modal maxima are combined as described in 3.7.2.1.5. The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses

resulting from any one horizontal or vertical excitation are considered to act simultaneously and the absolute values are added directly, as described in 3.7.2.6 and 3.7.2.7.

The relative displacements between anchors are determined from the dynamic analysis of the structures. All cases of relative displacement between anchors are considered. If significant, these relative displacements are then used in a static analysis to determine additional stresses imposed on equipment. Further details are given in 3.7.2.1.8.3.1 for the NSSS equipment and 3.7.3.9 for all other equipment. The cases where the relative displacements between anchors are insignificant and

Insert to Page 3.7-15

In the application of the alternate simplified method on WNP-2, a conservative static "g" loading was chosen for all piping systems when this approach was used irrespective of the building or building elevation. This simplifies the work and results in different amounts of conservatism for different piping systems. To confirm the adequacy of the alternate simplified method, a study is performed for several representative piping systems. Pipe stress and pipe support loads are calculated for these representative systems using response spectrum analysis methods. Results are examined to confirm that pipe stresses are within allowables and pipe support loads are less than these calculated using the simplified method.

Justification

A study was conducted to demonstrate that the equivalent static analysis criteria employed on WNP-2 is conservative as compared to response spectrum analysis methods.

A. Approach
Dynamic response spectra analysis was performed on the sample problems. The results from the equivalent static method and response spectrum method were compared.

B. Piping Systems Studied *

1. Six (6) Seismic Category I piping systems were selected as representative systems.
2. The piping systems chosen represent a variety of sizes from 3" to 24" pipe diameters.
3. Two systems were chosen from each of the Seismic Category I structures; the Reactor Building, the Service Water Pump House and the Diesel Generator Building.

C. Results of the Study
The results of the equivalent static analysis and the response spectrum analysis were compared for each piping system studied and is summarized as follows:

1. Where piping system design was established by the equivalent static analysis criteria, analyses using the response spectrum method have shown that stresses are well within code allowables in all cases.
2. A total of 192 pipe supports were compared. A summary of Support Loads for 30 of the 192 supports is shown in Table 1 and lists in ascending order the load ratio for the five (5) supports with the smallest load ratio in each system. The minimum ratio (load ratio) of equivalent static method/response spectrum method shown in Table 1 is 1.80.

D. Conclusion
The Equivalent Static Analysis Criteria used in piping analysis on the WNP-2 project is conservative and provides an adequate basis for piping system design.

* Additional considerations factored into the study to ensure that the six systems chosen for study represent a conservative basis for comparison:

(1) Piping systems located at higher elevations in the building were chosen for study, since seismic response spectra at higher elevations are larger, in order to represent a conservative basis for comparison.

(2) Piping systems which are extensive in length and thereby exhibiting a variety of configurations and spans were chosen.

TABLE 1 - RESULT OF STUDY Attachment 1

System	Location	Support Mark #	Calculated Load (lb)		Load Ratio ~~Equiv. Static Method / Response Spectrum Method~~
			Equiv.** Static Method	Response Spectrum Method	
Diesel Generator Air Intake	Diesel Generator Bldg.	DE-7 DE-7 DE-8 DE-14 DE-14	4696 4860 3120 4124 10505	2347 2281 1411 1780 2960	2.0 2.1 2.2 2.3 3.5
Loop B Return DG ENG 1B DMA-CC-21	Diesel Generator Bldg.	SW-261 SW-263 SW-253 SW-258 SW-257	854 535 1085 1128 873	173 107 151 145 112	4.9 5.0 7.2 7.8 7.8
Standby Service Water Pumphouse Spray Pond Crossover	Service Bldg.	SW-10 SW-11 SW-186 SW-187 SW-181	717 926 1109 1109 961	268 263 277 262 224	2.7 3.5 4.0 4.2 4.3
Standby Service Water Pumphouse Spray Pond Crossover	Service Bldg.	SW-1 SW-16 SW-17 SW-3 SW-14	579 563 1335 859 1429	330 286 527 326 445	1.80 2.0 2.5 2.6 3.2
Service Water from 20" SW Loop A	Reactor Bldg.	SW-322 SW-344 SW-324 SW-343 SW-321	359 253 683 426 490	178 67 147 83 87	2.0 3.8 4.7 5.1 5.6

** Design Basis

TABLE 1 - RESULT OF STUDY (continued)

System	Location	Support Mark #	Calculated Load (lb) Equiv.** Static Method	Response Spectrum Method	Load Ratio ~~Equiv. Static Method/ Response Spectrum Method~~
Service Water from 20" SW Loop B	Reactor Bldg.	SW-382	268	79	3.4
		SW-384	105	30	3.5
		SW-377	211	56	3.8
		SW-383	243	57	4.3
		SW-376	496	77	6.4

** Design Basis

DSER 3.7.3.b (MEB-8)

The applicant's procedures for the dynamic analysis of Category I systems, components, equipment and their supports have been reviewed by us and found to be generally acceptable. However, the following open issues must be resolved before we can report our findings.

a. Paragraph 3.7.2.1.8.2 (Page 3.7-15) stated that for the equivalent static load method, a minimum load factor of 1.15 is applied to building accelerations to include the effect of higher modes of vibration. The acceptance criteria of SRP 3.7.2 for the equivalent static load method is to apply a load factor of 1.5. A factor of less than 1.5 may be used if adequate justification is provided. Justification for utilizing this reduced factor is required.

b. Paragraph 3.7.3.2.1 of the FSAR states that "Based on Reference 3.7-10 (BWR/6 General Electric Standard Safety Analysis Report, Volume 1, General Electric Company, 4/30/74), which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event." Please provide clarification of this statement.

c. Paragraph 3.7.3.2.2 arrives at only one OBE intensity earthquake for design of the NSSS systems and components. Justification is required for this conclusion. Specifically, provide justification that the information in paragraph 3.7.3.2.2 is applicable to the WNP-2 site.

QUESTION NO. 8
(3.7.3)

Provide clarification of the statement in Paragraph 3.7.3.2.1 of the FSAR, "Based on Reference 3.7-10, which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE intensity. Therefore, the lifetime number of earthquake cycle may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event".

RESPONSE

See revised 3.7.3.2.1 of FSAR.

Summation - This item is closed.

K_j = Stiffness contribution of element j

W_i = Circular natural frequency of mode i

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

The general approach to the seismic subsystem analysis is identical to those procedures described in 3.7.2 for seismic system analysis, except for the soil/structure interaction effects.

3.7.3.1 Seismic Analysis Methods

The seismic analysis method used to analyze Seismic Category I subsystems is described in 3.7.2.1.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Number of Cycles for All Items Except NSSS Systems and Components

Assuming the mathematical model of strong motion earthquake acceleration described in 3.7.1.2 (T=15 seconds), the number of peaks and troughs, N, of the random process representing the structural response may be estimated. (Reference 3.7-3) The response of nuclear plant structures is controlled mostly by one governing mode, for the range of building frequencies normally encountered in nuclear plant facilities, (1.5 Hz to 6.0 Hz,) the number N is evaluated to be from 50 to 150. For a strong motion earthquake acceleration of 30 seconds in duration, N is from 100 to 300 for each seismic event. Fatigue evaluation due to a safe shutdown earthquake is not required by ASME Code, Section III since it qualifies as a faulted condition.

The operating basis earthquake is an upset condition and therefore must be included in fatigue evaluations according to ASME Code, Section III. The probability for the occurrence of a seismic event or OBE intensity is extremely low. Lower intensity earthquakes have a higher probability of occurrence. Based on Reference 10, which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered eqivalent to two earthquakes of OBE

As a minimum, 50 maximum stress cycles due to OBE are used for fatigue evaluations of BOP piping and components.

~~intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event.~~

~~During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.~~

3.7.3.2.2 Number of Cycles for NSSS Systems and Components

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec; (b) 1952, Taft N 69° W component, 30 sec; and (c) March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, (0^+-10 Hz, 10-20 Hz, and 20-50 Hz). This was done to provide a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior, as given in Table 3.7-18, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. This relationship is graphically shown in Figure 3.7-26.

DSER 3.7.3.c (MEB-9)

The applicant's procedures for the dynamic analysis of Category I systems, components, equipment and their supports have been reviewed by us and found to be generally acceptable. However, the following open issues must be resolved before we can report our findings.

a. Paragraph 3.7.2.1.8.2 (Page 3.7-15) stated that for the equivalent static load method, a minimum load factor of 1.15 is applied to building accelerations to include the effect of higher modes of vibration. The acceptance criteria of SRP 3.7.2 for the equivalent static load method is to apply a load factor of 1.5. A factor of less than 1.5 may be used if adequate justification is provided. Justification for utilizing this reduced factor is required.

b. Paragraph 3.7.3.2.1 of the FSAR states that "Based on Reference 3.7-10 (BWR/6 General Electric Standard Safety Analysis Report, Volume 1, General Electric Company, 4/30/74), which summarized data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event." Please provide clarification of this statement.

c. Paragraph 3.7.3.2.2 arrives at only one OBE intensity earthquake for design of the NSSS systems and components. Justification is required for this conclusion. Specifically, provide justification that the information in paragraph 3.7.3.2.2 is applicable to the WNP-2 site.

QUESTION NO. 9
(3.7.3)

Provide justification of utilizing one OBE intensity earthquake for design of the NSSS systems and components in Paragraph 3.7.3.2.2. Specifically, provide justification that the information in Paragraph 3.7.3.2.2. is applicable to the WNP-2 site.

RESPONSE

For the NSSS piping, 50 peak OBE cycles are used.

For other NSSS equipment and components, a generic study serves as the basis for 10 peak OBE cycles. As shown in the letter, R. Artigas to R. Bosnak, "Number of OBE Fatigue Cycles in the BWR NSSS Design", September 17, 1981, 10 peak OBE cycles can envelope the cumulative fatigue damage of hundreds of less severe earthquake cycles.

Accordingly, the FSAR is revised as attached.

Summation -
The applicant is to provide the comparison of the response spectra mentioned in the letter by November 20, 1981.*
This item is closed.

*The requested response spectra comparison is contained in the letter, R. Artigas to R. Bosnak, "Number of OBE Fatigue Cycles in the BWR NSSS Design," December 3, 1981 (attached).

9

intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event.

During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 3.7-10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level; and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.

3.7.3.2.2 Number of Cycles for NSSS Systems and Components

3.7.3.2.2.2 Other NSSS equipment and components

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec; (b) 1952, Taft N 69° W component, 30 sec; and (c) March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, (0^+-10 Hz, 10-20 Hz, and 20-50 Hz). This was done to provide a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior, as given in Table 3.7-18, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. ~~This relationship is graphically shown in Figure 3.7-26.~~

3.7.3.2.2.1 NSSS Piping

Fifty peak OBE cycles are postulated for fatigue evaluation.

9

In summary, the cyclic behavior number of fatigue cycles of a component during an earthquake is found in the following manner:

a. The fundamental frequency and peak seismic loads are found by a standard seismic analysis.

b. The number of cycles which the component experiences are found from Table 3.7-18 according to the frequency range within which the fundamental frequency lies.

c. For fatigue evaluation, one-half percent (0.005) of these cycles are conservatively assumed to be at the peak load and 4.5% (0.045) are assumed to be at or above three-quarter peak. The remainder of the cycles has negligible contribution to fatigue usage.

The safe shutdown earthquake has the highest level of response. However, the encounter probability of an SSE is so small that it is not necessary to postulate more than one SSE during the 40 year plant life. Fatigue evaluation due to the SSE is not necessary since it is a faulted condition and thus not required by ASME Code Section III.

The OBE is an upset condition and, therefore, must be included in fatigue evaluations according to ASME Code Section III. An investigation of seismic histories for many plants shows that during a 40 year life, it is probable that five earthquakes with intensities one-tenth of the SSE intensity, and one earthquake approximately 20% of the proposed SSE intensity, will occur. ~~Therefore, the probability of even an OBE is extremely low.~~ To cover the combined effects of these earthquakes and the cumulative effects of even lesser earthquakes, ~~one OBE intensity earthquake is~~ ten peak OBE cycles are postulated for fatigue evaluation.

Table 3.7-19 shows the calculated number of fatigue cycles and the number of fatigue cycles used in design.

3.7.3.3 Procedure Used for Modeling

The procedure used for modeling for the subsystem dynamic analysis is described in 3.7.2.3.

(9)

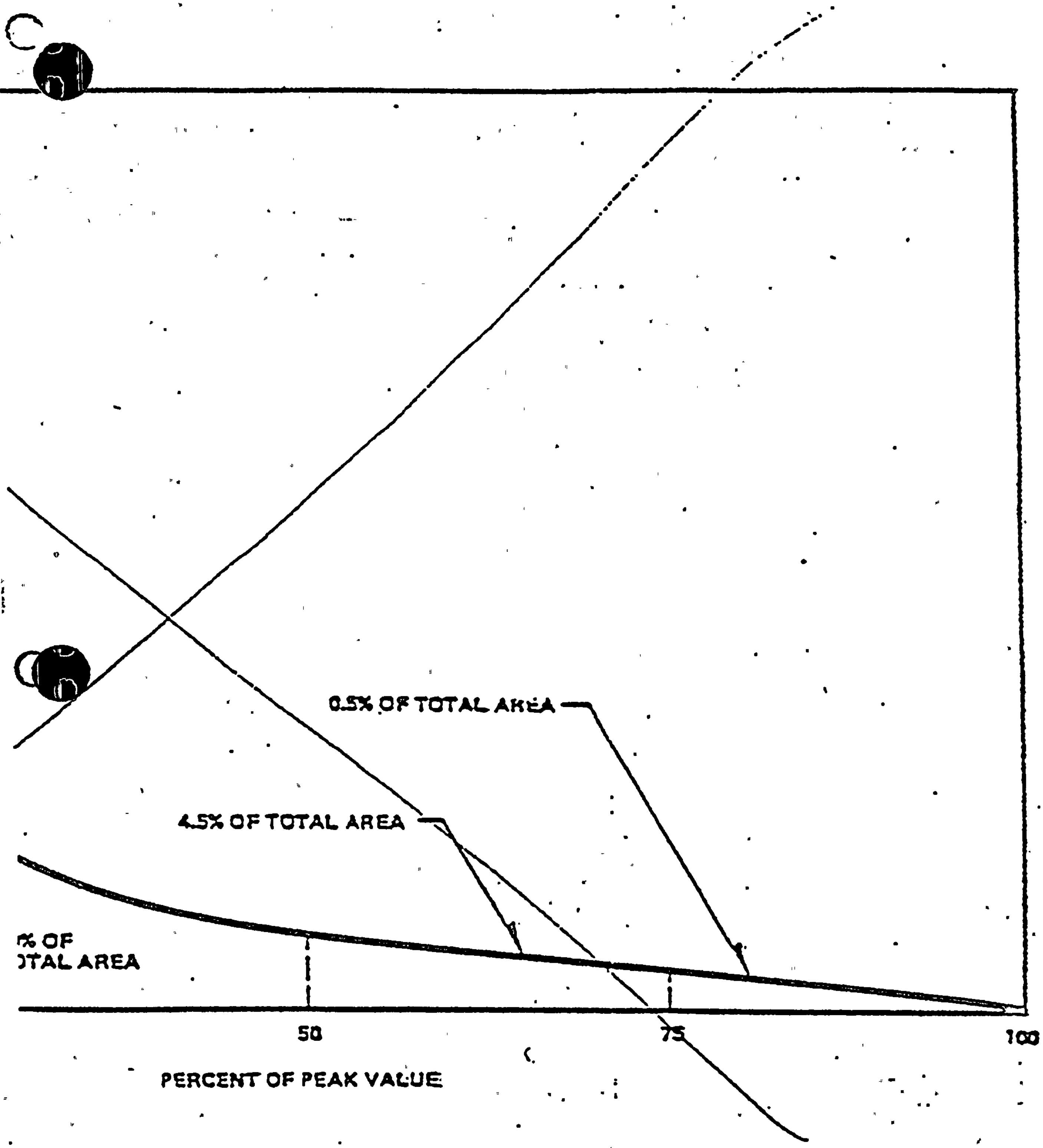
TABLE 3.7-18

NUMBER OF DYNAMIC RESPONSE CYCLES EXPECTED DURING A SEISMIC EVENT FOR NSSS SYSTEMS AND COMPONENTS

~~Frequency Band (H)~~ z	Frequency Bandwidth (Hertz)		
	0^{+} - 10	10 - 20	20 - 50
Total Number of Seismic Cycles	168	359	643
Seismic Cycles at Peak Load	0.8	1.8	3.2
Seismic Cycles at or above 75% of Peak Load	7.5	16.2	28.9

No. of seismic cycles (4.5% of total) between 50% and 75% of peak loads

No. of seismic cycles (0.5% of total) between 75% and 100% of peak loads



This figure is intentionally deleted.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	~~DENSITY OF STRESS REVERSALS~~	FIGURE 3.7-26

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

SYSTEMS DIVISION

MFN 171-81

September 17, 1981

Mr. R Bosnak, Chief
Mechanical Engineering Branch
Nuclear Regulatory Commission
Washington, D.C. 20555

Attachment to Q.9

Dear Mr. Bosnak

Subject: Number of OBE Fatigue Cycles in the BWR NSSS Design

This letter formally documents the meeting held on September 15, 1981 between the NRC and GE at Bethesda where GE presented the generic study in support of the use of 10 peak OBE cycles.

As we agreed at the end of the meeting, GE will take the following actions to close the related Clinton and Perry SER open items:

(1) For the NSSS piping, the FSAR will be amended to show that 50 peak OBE cycles were actually used in all piping calculations.

(2) For other NSSS equipment and components, the attached final version of the presentation package serves as the preliminary substantiation of the adequacy of 10 peak OBE cycles. This substantiation will be finalized when GE provides the comparison showing that the design basis response spectra of Clinton and Perry are bounded by those of Golden Gate, Taft, and El Centro earthquakes.

(3) The responses to the related MEB-SER questions will be rewritten to include the results of Actions (1) and (2) and resubmitted by November 6, 1981.

To resolve this same issues for other projects to be reviewed (including Hanford) by the NRC Staff, it is understood that the same approach will be taken to delineate the piping analysis and to justify the design adequacy of other equipment and components.

Your final concurrence is requested.

Very truly yours,

R. Artigas

R. Artigas, Manager
BWR Projects Licensing
Nuclear Safety & Licensing Operation
M/C 682, Ext. 53141

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125

SYSTEMS DIVISION

MFN 171-81

September 17, 1981

Attachment to Q.9

Mr. R Bosnak, Chief
Mechanical Engineering Branch
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Bosnak

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To resolve this same issues for other projects to be reviewed (including Hanford) by the NRC Staff, it is understood that the same approach will be taken to delineate the piping analysis and to justify the design adequacy of other equipment and components.

Your final concurrence is requested.

Very truly yours,

R. Artigas

R. Artigas, Manager
BWR Projects Licensing
Nuclear Safety & Licensing Operation
M/C 682, Ext. 53141

PRESENTATION ON THE NUMBER OF OBE FATIGUE CYCLES FOR BWR NSSS DESIGN (EXCEPT PIPING)

SEPTEMBER 15, 1981
D.K. HENRIE
SEISMIC & DYNAMIC ANALYSIS
GENERAL ELECTRIC

NUMBER OF OBE FATIGUE CYCLES

NSSS EQUIPMENT

- SRP RECOMMENDATION- 5 OBE WITH 10 CYCLES
- GE RECOMMENDATION- 10 PEAK OBE CYCLES GENERICALLY

GE STUDY SHOWS 10 PEAK OBE CYCLES OVER PLANT LIFE CONSERVATIVE

DKH-2

9/15/81

NRC NUREG/CR-1161 - RECOMMENDED REVISIONS
TO NRC SEISMIC DESIGN CRITERIA

- THE NUREG STATES THAT NRC REQUIREMENT "OF FIVE OBE CYCLES IS EXCESSIVELY CONSERVATIVE"

- ALSO INDICATES THAT ON THE AVERAGE, THE OBE DESIGN ACCELERATION HAS A NEP OF 90% IN A 50 YEAR LIFE

- WASH-1400, OCTOBER 1975 - PROBABILITY OF OBE IS ONE IN 100 TO 125 YEARS AND NOT FIVE IN 40 YEARS.

CONCLUSION -	5 OBE EXCESSIVELY CONSERVATIVE

DKH-3

9/15/81

GE STUDY ON PROBABILITY OF OBE (1973)

- o BASIS - A STUDY OF 26 PSAR AND FSAR PLANTS
- o FOUR 40 YEAR PERIODS
  - 1810 - 1849
  - 1850 - 1889
  - 1890 - 1929
  - 1930 - 1969
- o MAXIMUM SITE INTENSITY EARTHQUAKE CHOSEN FROM EACH PERIOD FOR EACH SITE
- o RATIO OF MAXIMUM GROUND ACCELERATION TO SSE DESIGN BASIS GROUND ACCELERATION CALCULATED FOR EACH 40 YEAR PERIOD
  - MAXIMUM = 0.16
  - MINIMUM = INSIGNIFICANT
  - MEAN = 0.051
  - STANDARD DEVIATION= 0.039

JKP- 4
9/15/81

TAB[illegible]

GENERIC SUMMARY

	EL CENTRO	TAFT	GOLDEN GATE	CLINTON	HANFORD	PERRY	SUSQUEHANNA
DURATION (SEC.)	29.4	30.0	13.2	10.0	16.0	10.0	15.0
MAX. SITE ACCEL.(g) (RECORDED/ESTIMATED)	0.33	0.18	0.13	.015	.015	.007	.007
SSE DESIGN BASIS MAX. ACCEL. (g)	-	-	-	.25	.25	.15	.10
MAX. HORIZ. ACCEL (g)/ 90% NEP IN 50 YEARS	-	-	-	<0.04	<0.10	0.07	<0.04

DKH-5

9/15/81

- NUMBER OF FATIGUE CYCLES PER EARTHQUAKE
  - OBTAINED BY TIME HISTORY ANALYSIS
  - RANDOM VS. PERIODIC EXCITATION
  - TABLE 2 - ENVELOPED AND AVERAGED CYCLES FROM THREE EARTHQUAKES AND SIX MAJOR NSSS COMPONENTS
  - TABLE 3 - % OF CYCLES < 50% OF PEAK
  - TABLE 4 - % OF CYCLES < 25% OF PEAK
  - INDEPENDENT OF EARTHQUAKE OR COMPONENT FREQUENCY
    - 99.5% OF STRESS REVERSALS OCCUR BELOW 75% OF MAXIMUM STRESS
    - 95% BELOW 50%
    - 85% BELOW 25%
  - TABLE 5 - SUMMARY OF EQUIVALENT STRESS CYCLES OF ALL MAGNITUDES

CONCLUSION - 10 PEAK OBE CYCLES ARE CONSERVATIVE

DKH-6
9/15/81

TAB

AVERAGE NUMBER OF STRESS CYCLES OF ALL MAGNITUDES

LONG DURATION EARTHQUAKE	DURATION (SEC)	PEAK ACCEL. (g)	NORMALIZED PEAK ACCEL. (g)	NUMBER OF CYCLES FREQUENCY BANDS 0-10 Hz	10 - 20 Hz	20-50 Hz
EL CENTRO(1)	29.4	0.33	0.25	$\boxed{168}$	337	425
TAFT(2)	30.0	0.18	0.25	163	$\boxed{359}$	$\boxed{643}$
GOLDEN GATE(3)	13.2	0.13	0.25	94	171	316

NOTES:
(1) May 18, 1940, El Centro, N/S Component, 29.4 sec
(2) July 21, 1952, Taft, S69°E Component, 30.0 sec.
(3) March 22, 1952, Golden Gate, S80°E Component, 13.2 sec.

DKH-7
9/15/81

TA[illegible]8

PERCENTAGE OF STRESS CYCLES
WITH STRESS AMPLITUDES BELOW 50% OF THE MAXIMUM VALUE

Frequency Range	0 - 10 HZ			10 - 20 Hz			20 - 50 Hz		
Earthquake	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate
Duration, sec	29.4	30.0	13.2	29.4	30.0	13.2	29.4	30.0	13.2
Component				P E R C E N T A G E S					
A (STABILIZER)	99.2	98.0	99.9	94.7	95.6	90.8	94.7	99.2	96.8
B (STAR TRUSS)	99.2	97.1	99.9	97.9	97.9	96.8	97.6	99.[illegible]	97.7
C (REACTOR SKIRT)	99.0	95.8	99.9	95.4	96.4	92.8	99.6	99.8	99.3
D (SHROUD SUPPORT)	99.1	96.9	99.5	96.2	97.2	94.1	99.2	99.6	98.9
E (FUEL)	99.1	95.8	99.9	95.3	96.8	91.3	95.7	99.8	99.1
F (CRD HOUSING)	98.5	95.7	99.4	96.6	96.5	94.4	95.6	98.6	97.9
Average (Over-All Average 97.4)	99.0	96.6	99.8	96.0	96.7	93.4	97.1	99.4	98.3

Time History Input Cycles Below 50% of Peak

El Centro	93%
Taft	90%
Golden Gate	95%

DKH-8
9/15/81

TABLE 4

PERCENTAGE OF STRESS CYCLES
WITH STRESS AMPLITUDES BELOW 25% OF THE MAXIMUM VALUE

Frequency Range	0 - 10 Hz			10 - 20 Hz			20 - 50 Hz		
Earthquake	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate	El Centro	Taft	Golden Gate
Duration, sec	29.4	30.0	13.2	29.4	30.0	13.2	29.4	30.0	13.2
Component	PERCENTAGE								
A	85.5	85.4	96.2	80.6	77.6	81.3	81.9	89.4	87.5
B	85.8	84.9	96.2	80.4	78.0	80.6	82.6	80.3	89.3
C	93.1	81.2	99.4	82.3	77.3	81.9	92.1	92.8	94.0
D	86.0	81.0	98.0	84.3	79.6	83.4	90.0	93.0	91.4
E	89.4	79.4	97.6	82.0	79.1	80.6	80.9	91.2	90.4
F	86.6	79.5	92.2	84.4	78.9	83.3	79.5	89.0	89.0
Average	87.7	81.9	96.6	82.3	78.4	81.9	84.5	89.3	90.3
(Over-all average	85.9)								

Time History Input Cycles Below 25% of Peak

El Centro	78%
Taft	70%
Golden Gate	90%

TA[illegible]5

NUMBER OF STRESS CYCLES OF ALL MAGNITUDES
DURING A LONG DURATION EARTHQUAKE

	FREQUENCY BANDS (CORRESPONDS TO COMPONENT FUNDAMENTAL FREQUENCIES)					
	0 - 10 Hz		10 - 20 Hz		20 - 50 Hz	
TOTAL NUMBER OF STRESS CYCLES	168		359		643	
NUMBER OF CYCLES BETWEEN 75% AND 100% OF PEAK VALUE (0.5% OF TOTAL)	1	(1)	2	(2)	3	(3)
NUMBER OF CYCLES BETWEEN 50% AND 75% OF PEAK VALUE (4.5% OF TOTAL)	8	(1)	16	(2)	29	(4)
NUMBER OF CYCLES BETWEEN 25% and 50% OF PEAK VALUE (10% OF TOTAL)	17	(1)	36	(1)	64	(1)
NUMBER OF CYCLES LESS THAN OR EQUAL TO 25% OF PEAK VALUE (85% OF TOTAL)	143	(1)	305	(1)	547	(1)
TOTAL NUMBER OF EQUIVALENT PEAK STRESS CYCLES		(4)		(6)		(9)

DKH-10

9/15/81

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-3141

December 3, 1981

Mr. R. J. Bosnak, Chief
Mechanical Engineering Branch
Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Bosnak:

SUBJECT: NUMBER OF OBE FATIGUE CYCLES IN THE BWR NSSS DESIGN

References:
1) Letter, Bosnak to Artigas, same subject, dated December 2, 1981
2) Letter, Artigas to Bosnak, same subject, dated September 17, 1981

Reference 1 has defined a reasonable and justifiable framework within which to close the subject issue for all projects subject to OL review. This letter reiterates GE's position on this issue as well as provides the information requested by the NRC for all projects pending the OL issuance.

A criterion acceptable to the NRC and GE providing the basis for 10 peak OBE cycles in the BWR NSSS design has been established as a result of the September 15 meeting and subsequent telephone discussions. As documented in References 1 and 2, the justification of this criterion is summarized as follows:

1) In a base study, GE subjected a typical BWR to three historically recorded earthquakes - El Centro, Taft, and Golden Gate - of severe magnitude and long duration. This study demonstrated the adequacy of ten peak OBE cycles for fatigue evaluation.

2) If it can be shown that a project unique design base OBE is bounded by the three base study earthquakes, then the likely number of peak OBE cycles required to account for the cumulative stress damage in the plant life must be less than ten.

3) Magnitude, duration, and response spectrum comparisons in combination can show that the design basis OBEs are bounded by the base study earthquakes. The sample response spectrum comparison for Clinton, Perry, and Hanford enclosed in Attachment 1 and the magnitude and duration comparisons in Reference 2 have demonstrated

that the base study earthquakes do indeed bound the site-specific design basis OBEs.

Therefore, the use of ten peak OBE cycles is conservative for all project unique designs. This GE position is supported by NUREG/CR-1161, which recommends revisions to the NRC seismic design criteria. The NUREG cites that (a) the requirement of 50 OBE cycles is excessively conservative, and (b) the probability of 10 OBE cycles (as a result of one OBE) is one in 100 to 125 years. In conclusion, GE firmly believes this current approach is technically sound.

However, to address the Staff's concern and to expeditiously resolve this licensing issue, GE intends to demonstrate the achievement of an equivalent level of safety by showing the relevant design margin. Specifically, GE has agreed to substantiate that, for the most limiting RPV component, (a) margin is allowed in the total cumulative fatigue usage, and (b) the contribution of OBE cycles is very negligible in this total usage.

Referring to Attachment 2, the usage factor for the most limiting component, feedwater nozzle, is tabulated for BWR/5's and 6's. For BWR/5's, the highest usage factor occurs on the Hanford 2 project. This precludes the need to list other projects. For BWR/6's, the listed values are used for generic design. Both cases show that the contribution of OBEs is very insignificant and adequate margin is built into the design.

This completes our commitment and response to your inquiry. We understand that this issue can be written off for all BWR/5 and 6 Projects based on this submittal.

Very truly yours,

R. Artigas

R. Artigas, Manager
BWR Projects Licensing
Nuclear Safety and Licensing Operation

RA:hmc:sem/107J

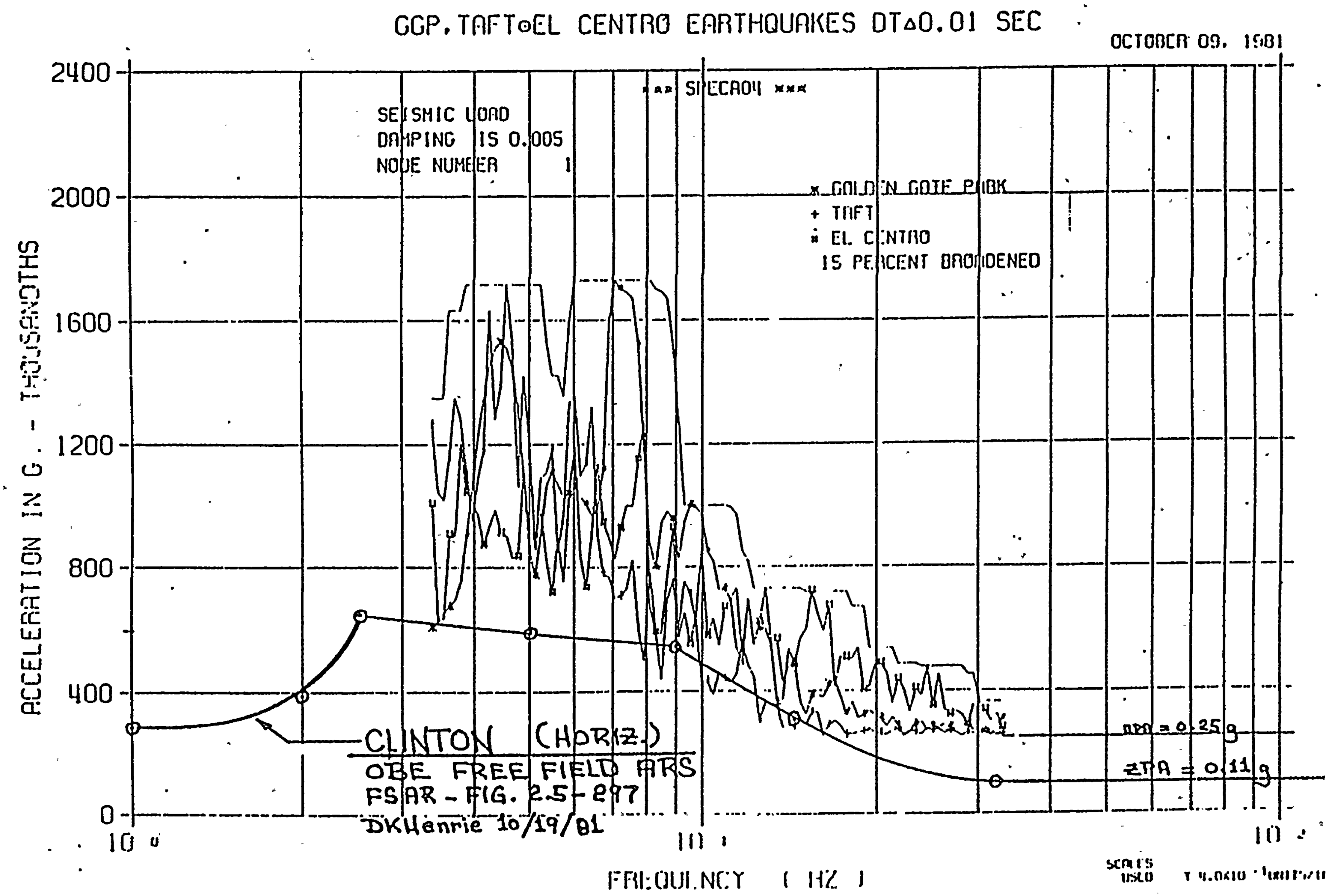
Attachments

<u>Attachment 1</u>

OBE Response Spectrum Comparisons

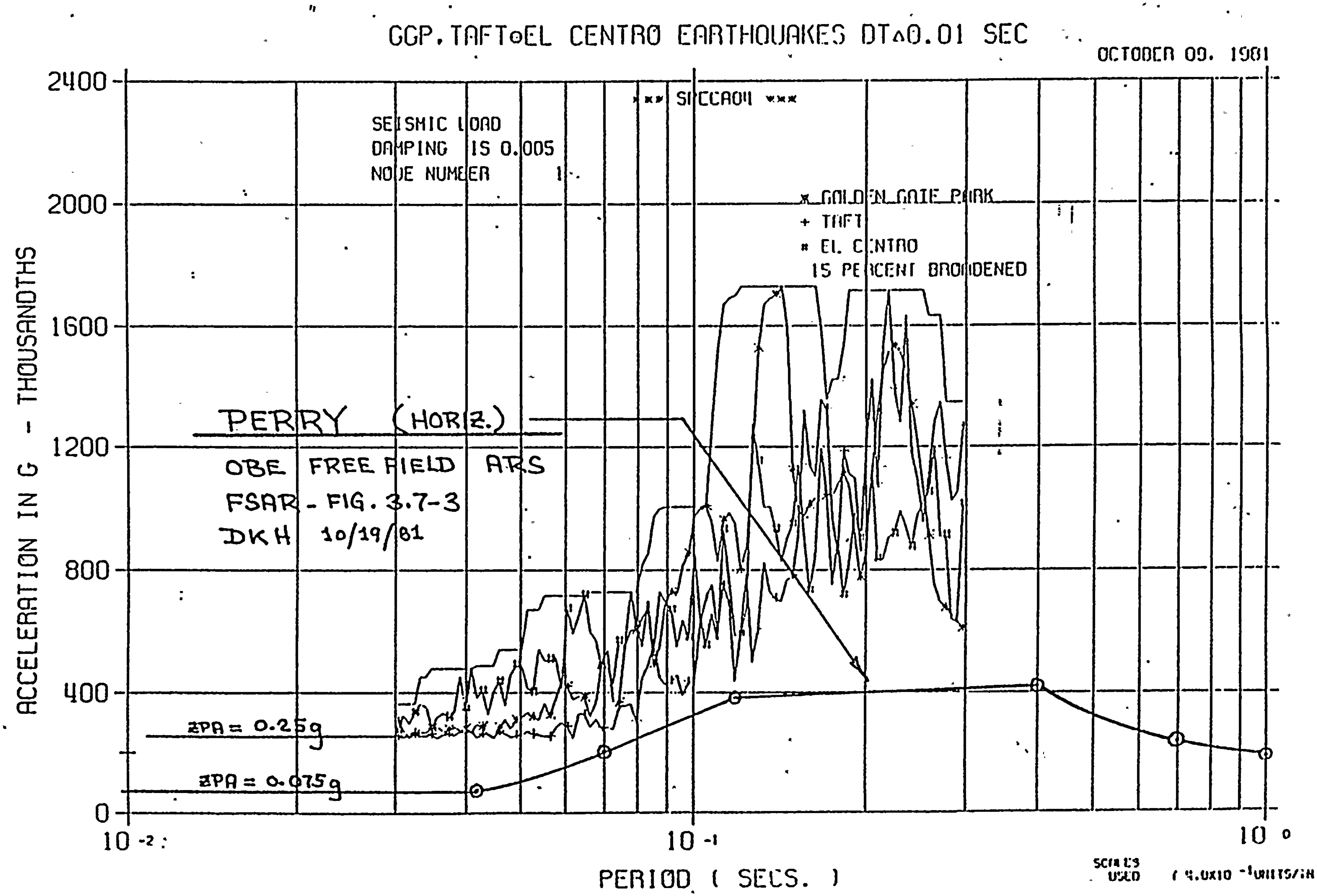
Clinton, Perry, and Hanford vs. Base Study Earthquakes

PCY:sem/2B
12/1/81



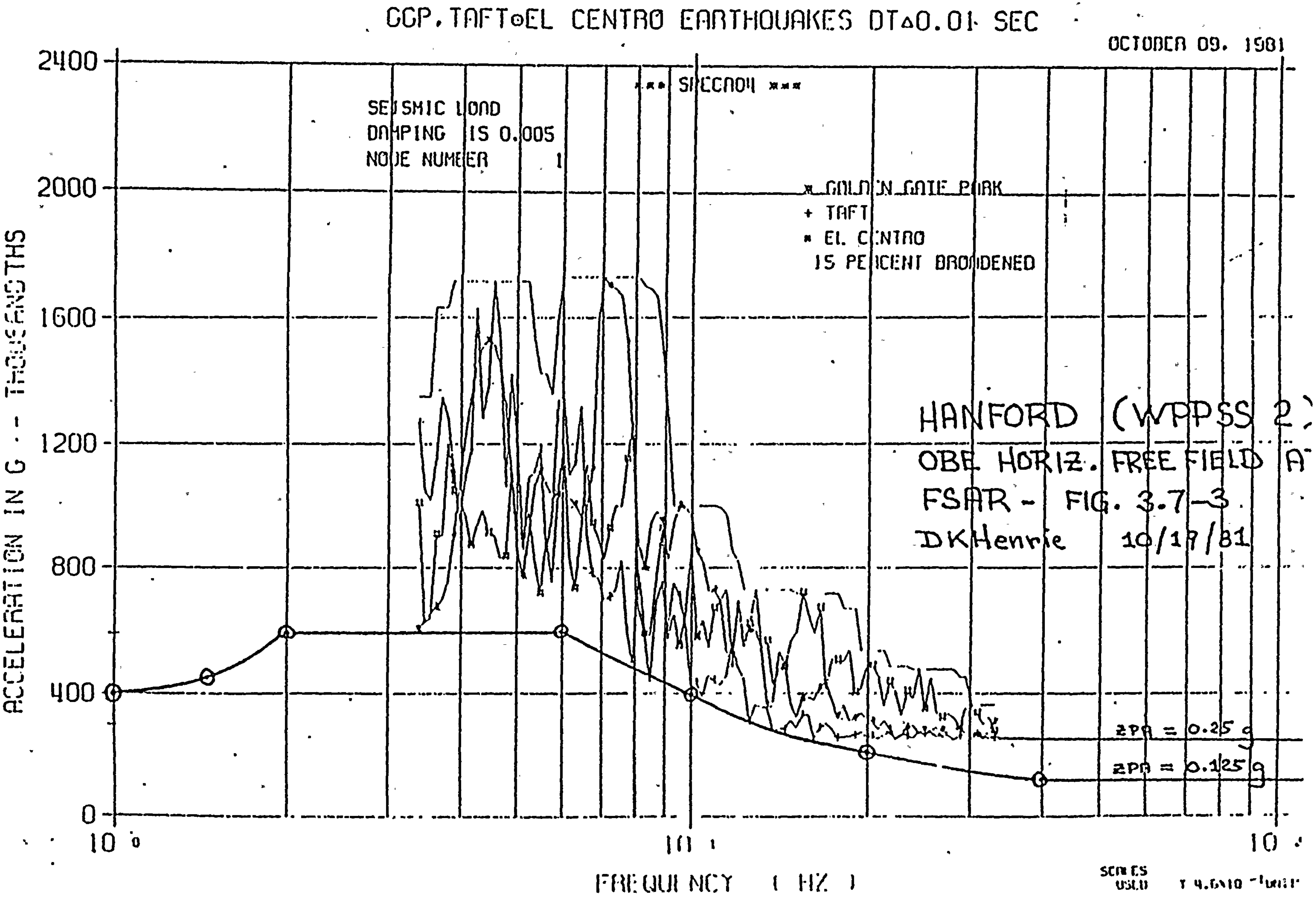
GGP, TAFT & EL CENTRO EARTHQUAKES DT△0.01 SEC
OCTOBER 09. 1981
*** SPECA04 ***
SEISMIC LOAD
DAMPING IS 0.005
NODE NUMBER 1
× GOLDEN GATE PARK
+ TAFT
× EL CENTRO
15 PERCENT BROADENED
ACCELERATION IN G. - THOUSANDTHS
2400
2000
1600
1200
800
400
0
FREQUENCY (HZ)
CLINTON (HORIZ.)
OBE FREE FIELD ARS
FSAR - FIG. 2.5-297
DKHenrie 10/19/81
ZPA = 0.25 g
ZPA = 0.11 g

PL 1



GGP, TAFT & EL CENTRO EARTHQUAKES DT=0.01 SEC
OCTOBER 09, 1981
*** SPECTRUM ***
SEISMIC LOAD
DAMPING IS 0.005
NODE NUMBER 1
× GOLDEN GATE PARK
+ TAFT
# EL CENTRO
15 PERCENT BROADENED
PERRY (HORIZ.)
OBE FREE FIELD ARS
FSAR - FIG. 3.7-3
DKH 10/19/81
ZPA = 0.25g
ZPA = 0.075g
ACCELERATION IN G - THOUSANDTHS
2400
2000
1600
1200
800
400
0
10 -2
10 -1
10 0
PERIOD (SECS.)
SCALES USED

PL 1


GGP, TAFT & EL CENTRO EARTHQUAKES DT=0.01 SEC
OCTOBER 09, 1981
*** SPECCNOH ***
SEISMIC LOAD
DAMPING IS 0.005
NODE NUMBER 1
× GOLDEN GATE PARK
+ TAFT
* EL CENTRO
15 PERCENT BROADENED
HANFORD (WPPSS 2)
OBE HORIZ. FREE FIELD A
FSAR - FIG. 3.7-3
DKHenrie 10/19/81
ZPA = 0.25 g
ZPA = 0.125 g
ACCELERATION IN G - THOUSANDTHS
FREQUENCY (HZ)
2400
2000
1600
1200
800
400
0

PI 1

Attachment 2

Fatigue Usage of RPV Feedwater Nozzle

Loading	BWR/4[2]	BWR/5[3]	BWR/6[4]
10 OBE Cycles	Later	<0.001	0.006
All Others[1]	Later	0.966	0.944
Total	Later	0.966	0.950

(1) All other fatigue contributions due to SRV, thermal, operating transients, etc..
(2) To be provided later when final calculations are completed for the Limerick or Hope Creek project.
(3) WNP-2 (Hanford) has the highest usage among BWR/5's.
(4) Generic design numbers used for all BWR/6's.

PCY:sem/2C
12/3/81

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

RECEIVED
DEC 2 1981
P.C. Yin

Mr. R. Artigas, Manager
BWR Projects Licensing
Nuclear Safety & Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Reference: Letter from R. Artigas to R. Bosnak dated
September 17, 1981 with attachment.

Dear Mr. Artigas:

In the above referenced letter, GE documented its presentation given to the NRC at the GE Bethesda offices on September 15, 1981. The GE presentation provided the basis for using 10 peak OBE cycles for the seismic fatigue design of BWR NSSS components. Currently, the Standard Review Plan accepts 5 OBE's with a minimum of 10 cycles per earthquake. The GE presentation was based on studies performed by GE using the time-histories of the Taft, Golden Gate, and EL Centro earthquakes.

During the above meeting, the NRC expressed concern that the design basis or site-specific earthquakes applicable to each BWR plant might not be bounded by the three earthquakes used in the GE studies. GE proposed, in the above referenced letter, to provide a comparison showing that the design basis seismic response spectra for the Clinton, Perry and WPPSS-2 plants are bounded by the response spectra of the three earthquakes used in the GE presentation. Subsequently, we discussed with GE our position that a response spectra comparison may not provide a complete justification for the use of 10 peak OBE cycles because the GE presentation only provides an adequate basis for assuming 10 equivalent peak OBE cycles for those plants whose site-specific earthquakes are bounded by the Taft, Golden Gate, and El Centro earthquakes. Consequently, our review of each plant's geological and seismological characteristics might result in the use of more than 10 peak cycles for a site-specific earthquake. Thus, based on our evaluation of the GE presentation alone, we were not able to substantiate the use of 10 peak plant-specific OBE cycles for fatigue design.

After several discussions with GE, it was clarified that in the generic BWR 6 component fatigue design, bounding values were assumed for the 10 peak OBE cycles. These bounding values were based on a conservative OBE and typical plant-specific OBE values are much less than the OBE values assumed in the bounding generic design.

It is our understanding that for each BWR 6 plant, analyses of the RPV components are performed using plant-specific data to confirm the adequacy of the generic design. Similarly, conservative design margins also exist for BWR 4 and BWR 5 plants. GE has stated that they will provide for each BWR 4, 5 and 6 plant undergoing OL review, a table showing (1) the total cumulative usage factor for the most limiting RPV component and (2) the contribution of the 10 peak OBE cycles to that fatigue cumulative usage factor. We concur with the above GE commitments.

In the above referenced letter, GE stated that for the fatigue evaluation of NSSS piping systems, 50 peak OBE cycles are used. This is in accordance with the Standard Review Plan and, thus, we find it to be acceptable.

In conclusion, based on an acceptable review of each plant's geological and seismological characteristics, it is the staff's position that the use of 10 bounding design OBE peak cycles is acceptable for the generic fatigue design of the RPV components. However, for each BWR 4, 5 and 6 plant undergoing OL review, the staff will review the total cumulative usage factor for the most limiting RPV component and the seismic contribution to the cumulative usage factor to determine whether there exists a level of safety equivalent to that provided in the Standard Review Plan. If our review of the cumulative usage factor cannot provide assurance that an acceptable level of safety exists, then we will require further justification from GE. The results of our review will be included in the Safety Evaluation Report of each plant undergoing operating license review.

R. J. Bosnak

Robert J. Bosnak, Chief
Mechanical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

DSER 3.8.2(SEB-1)
DSER 3.8.2(a), (b), and (c)

that is, plants in the later stage of construction, the staff issued NUREG-0487 report entitled "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," dated October 1978, with Supplements 1 and 2 dated August 1980 and January 1981, respectively.. In August 1981, the regulatory staff issued NUREG-0808 report entitled, "Mark II Containment Program Load Evaluation and Acceptance Criteria" in which the staff concludes that the improved condensation-oscillation and chugging loads for the suppression pool boundary as proposed by the Mark II Owners' Group and the lead plant pool-swell loads adopted by the Mark II owners as the final load specifications are conservative. In order to meet the requirements of the SRP Section 3.8.2.II.4d which states that an analysis should be performed to determine the ultimate capacity of the containment, the applicant will submit information of such an analysis by November 13, 1981 for staff review and acceptance.

On the basis of DFFR and NUREG-0487, the concrete structural components forming the boundary of the suppression pool were evaluated by the applicant for their capability to resist the effects of the additional hydrodynamic loads and were found to have adequate margins of safety. Through the use of a finite element model with the inclusion of the water as fluid mass, the effect of fluid-structure interaction was considered in the evaluation. The evaluation is contained in the applicant's Design Assessment Report (DAR). The staff has reviewed the DAR and found additional information in the following areas is required: (a) analysis of fatigue for the steel containment shell, (b) the effects of shell stiffening and opening on the applicability of ASME Code, Section III, NE-3133 for buckling analysis, and (c) the design of the concrete above and below the steel containment bottom head. As soon as we receive the above required information and have reviewed and found it to be satisfactory, we shall be in a position to concur with the applicant's conclusion. Additionally, the Revision 2 to the DAR was completed in August 1979. Since then more information has been generated from the Mark II Generic Program. Therefore, the criteria used in the evaluation are not totally in conformance with those delineated in NUREG-0808. In view of this fact, the applicant has committed to make an assessment of WNP-2 containment with respect to the effect of revisions to load definitions as delineated in NUREG-0808 as a confirmation of the adequacy of the evaluation as presented in the August 1979 DAR.

The following conclusion is subject to the satisfactory resolution of the above unresolved items.

Open SER Issue

3.8.2 Steel Containment Ultimate Capacity (SEB-1)

A response to this issue was submitted December 14, 1981 by letter number G02-81-518.

3.8.2(a) Containment Shell Fatigue Analysis

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

3.8.2(b) Effects of Shell Stiffening and Opening

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

3.8.2(c) Design of Concrete Above/Below Head

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

DSER 3.8.3(a) and (b)

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads, including pressure and jet loads, and seismic loads. The design of the containment internal structures to withstand the effects of suppression pool hydrodynamic loads was accomplished in the same manner as that for the containment structure as described in Section 3.8.2. The detailed reevaluation of the capability of containment internal structures to resist these newly identified loads is described in the applicants' Design Assessment Report. We have reviewed the design and analysis procedures and criteria that were used for the original design and for the reevaluation of the internal structures in the suppression pool. The containment internal structures were designed and proportioned to remain within limits established by the Regulatory staff under various load combinations. These limits as well as the design and analysis procedures are, in general, based on the American Concrete Institute 318-71 Code and on the American Institute of Steel Construction Specification for Concrete and Steel Structures, respectively, modified as appropriate for load combinations that are considered as extreme.

The applicant will provide impact assessment to demonstrate compliance with ACI-349 Code as augmented by Regulatory Guide 1.142 for the applicable internal concrete structures and also to demonstrate compliance with ASME Code, Section III, Division 2 for the containment concrete basemat structures.

The loads and load combinations used in Tables 3.8-10 and 3.8-11 in Section 3.8.3 of the FSAR are different from those of Section 3.8.4 of the SRP. However, the applicant provided reevaluation design calculations and showed that the structures internal to containment have enough capacity to meet the applicable requirements of Section 3.8.4 of the SRP. The staff reviewed the results of the reevaluation and accepted the applicant's justification.

The following conclusion is subject to the resolution of the unresolved items discussed in this section.

The materials of construction, their fabrication, construction, and installation, are in accordance with the American Concrete Institute 318-71 Code as

Open SER Issue

3.8.3(a) Internal Structure Compliance with ACI-349/RG1.142

A response to this issue was submitted December 14, 1981 by letter number G02-81-518.

3.8.3(b) Basemat Compliance with ASME Code

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

DSER 3.8.4-(a)

The design and analysis procedures that were used for these Category I structures are the same as those approved on previously licensed applications and in general, are in accordance with procedures delineated in the ACI 318-71 Code and in the AISC Specification for concrete and steel structures, respectively.

The various Category I structures are designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction, and installation, are in accordance with the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

The applicant committed to submit assessment report to demonstrate that the design of Category I concrete structures is in compliance with the requirements of ACI-349 Code as amended by Regulatory Guide 1.142.

The loads and load combinations used in Table 3.8-15 and 3.8-16 in Section 3.8.4 of the FSAR are different from those presented in Section 3.8.4 of the SRP. The applicant has reevaluated the design of WNP-2 and indicated that the load combinations and acceptance criteria specified in Section 3.8.4 of the SRP are satisfied. This is acceptable to the staff.

The applicant committed to submit an evaluation of the design and analysis of spent fuel pool structures used in WNP-2 for staff review. A copy of "Minimum Requirements for Design of Spent Fuel Pool Racks" has been given to the applicant.

The applicant confirmed that there are no safety-related masonry walls for WNP-2 facility.

The following conclusion is subject to the satisfactory resolution of the above noted unresolved items.

Open SER Issue

3.6.4(a) Category I Structures' Compliance with ACI-349/RG1.142(SEB-11)

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

DSER 3.8.4(b)

The design and analysis procedures that were used for these Category I structures are the same as those approved on previously licensed applications and in general, are in accordance with procedures delineated in the ACI 318-71 Code and in the AISC Specification for concrete and steel structures, respectively.

The various Category I structures are designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC Specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction, and installation, are in accordance with the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

The applicant committed to submit assessment report to demonstrate that the design of Category I concrete structures is in compliance with the requirements of ACI-349 Code as amended by Regulatory Guide 1.142.

The loads and load combinations used in Table 3.8-15 and 3.8-16 in Section 3.8.4 of the FSAR are different from those presented in Section 3.8.4 of the SRP. The applicant has reevaluated the design of WNP-2 and indicated that the load combinations and acceptance criteria specified in Section 3.8.4 of the SRP are satisfied. This is acceptable to the staff.

The applicant committed to submit an evaluation of the design and analysis of spent fuel pool structures used in WNP-2 for staff review. A copy of "Minimum Requirements for Design of Spent Fuel Pool Racks" has been given to the applicant.

The applicant confirmed that there are no safety-related masonry walls for WNP-2 facility.

The following conclusion is subject to the satisfactory resolution of the above noted unresolved items.

Open SER Issue

3.8.4(b) Evaluation of Spent Fuel Pool Structures

The WNP-2 spent fuel pool racks' design meets the requirements of the NRC guidelines promulgated by the NRC letter to all Power Reactor Licensees with enclosure, "Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, (supplemented January 1979).

Attached is a comparison of the April 14, 1978 guidelines to those of Appendix D to NUREG-0800 Standard Review Plan Section 3.8.3, Revision 0, dated July 1981.

ATTACHMENT A

Review of Standard Review Plan 3.8.4 Appendix D Requirements

Appendix D of Standard Review Plan Section 3.8.4 "Technical Position on Spent Fuel Pool Racks" was reviewed and compared to the existing NRC guidelines made available to all power reactor licenses in April 1978 and then modified in January 1979.

The new appendix addresses only the mechanical, material, and structural considerations; nuclear and thermal-hydraulic considerations are not included. A comparison is given below for the subject areas covered by both documents.

Appendix D Section (1) (b) Fuel Handling

Where specific postulated drop accidents had previously been defined, Appendix D implies that accident postulation should be made by the licensee and then submitted to the NRC for review by the Accident Evaluation Branch. The fuel handling and postulated load drop analysis performed for WNP-2 design are considered sufficient and should meet the requirements of such a review.

Appendix D Section (2) Applicable Codes, Standards and Specifications

Appendix D Section 2 states that design, fabrication and installation of spent fuel racks of stainless steel material may be performed based upon Subsection NF requirements of the ASME Section III Code. This implies that another code may also be used, however, in Section (6) (the structural acceptance criteria section) Subsection NF is imposed. This leads to confusion.

The original guidelines permitted the AISC Code (supplemented by Standard Review Plan 3.8.4-II-5) to be used as an alternate design criteria. This alternate was the criteria used on the WNP-2 spent fuel rack design.

Appendix D Section (3) Seismic and Impact Loads

Appendix D permits the effects of submergence of the rack system to be considered on a case by case basis. In the original guidelines the added mass of the water had to be considered (amplifies loads) while increased damping due to submergence (reduces loads) could not be applied without test data and/or detailed analytical results. Consequently, the original guidelines was more restrictive, leading to larger resultant loads in the case of dynamic loading conditions since water damping was not permitted.

Appendix D Section (4) Loads and Load Combinations

Appendix D requires that the racks, pool slab, and fuel pool liner be evaluated for accident load combinations which include the impact of the spent fuel cask. In the original guidelines, acceptance is based on conforming with the applicable portion of Section 3.8.4-II-3 which does not include an impact loading condition of a dropping cask. Such a load is typically dismissed if redundant lifting capabilities and locks are provided with overhead crane equipment. Such is the case on the WNP-2 crane design.

The original guidelines imposed Section 3.8.4 load combinations when analysis was performed in accordance with AISC. A table similiar to table I was provided with load combinations and less restrictive acceptance criteria limits, when analyzing to Section III, Subsection NF of the ASME Code.

Appendix D Section (5) Design and Analysis Procedures

There are no changes in the section when compared to the original guidelines

Appendix D Section (6) Structural Acceptance Criteria

This section imposes a given set of load combinations (Table 1) and acceptance criteria with reference only to Section III Subsection NF and Appendix XVII. The option to choose between the AISC and ASME Codes has been removed. The attached Table A provides a review of critical load combinations and acceptance limits between the original and revised guidelines of 1978 and 1979 and Appendix D.

Appendix D Section (7) Material, Quality Control and Special Construction Techniques

Methods for structural qualification of poison material and test of hardened stainless steel material is no longer covered in this section. Section (8) of the original guidelines "Testing and Inservice Surveillance" is not included in Appendix D.

TABLE A

A COMPARISON OF THE LOAD COMBINATION AND ACCEPTANCE CRITERIA FOR THE DESIGN OF SPENT FUEL RACKS

Load Combinations	April 1978 Reference 1 Acceptance Criteria Original Guidance AISC	ASME	January 1979 Reference 2 Acceptance Criteria Modified Guidance AISC	ASME	July 1981 Reference 3 Acceptance Criteria Appendix D ASME (only)
1. D+L	1.0S	Normal Limit of NF 3231.1a = 1.0S	1.0S	Normal Limits of NF 3232.1a = 1.0S	1.0S
2. D+L+E	1.0S	Normal Limits of NF 3231.1a = 1.0S	1.0S	Normal Limits of NF 3231.1a = 1.0S	N/A
3. $D+L+T_o$	1.5S	1.5S or lesser of 2Sy of Su	1.5S	Lesser of 2Sy or Su stress range	1.0S
4. $D+L+T_o+E$	1.5S	1.5S or lesser of 2Sy of Su	1.5S	Lesser of 2Sy or Su stress range	1.0S
5. $D+L+T_a+E$	1.6S(3)	1.6S or lesser of 2 Sy or Su	1.6S(3)	Lesser of 2Sy or Su stress range	1.0S
6. $D+L+T_o+E'$	1.6S(3)	N/A	1.6S(3)	N/A	N/A
7. $D+L+T_o+P_f$	-(1)	N/A	-(1)	N/A	1.0S
8. $D+L+T_a+E'$	1.7S(3)	Faulted conditons limits of NF 3231.1c = lesser of 1.2Sy or 0.75Su(2)	1.7S(3)	Faulted condtion limits of NF 3231.c = lesser of 1.2Sy or 0.7Su(2)	Faulted condition limits of NF 3231.1c = lesser of 1.2Sy or 0.7Su(2)
9. $D+L+F_d$	Functional capability of fuel rack should be demonstrated	Functional capability of fuel rack should be demonstrated	Functional capability of fuel rack should be demonstrated	Functional capability of fuel rack should be demonstrated	Functional capability of fuel rack should be demonstrated(4)

Notes:

(1) Not included in load combinations but considered and generally compared to 1.6S

(2) Thermal stresses need not be considered

(3) Thermal loads can, in general, be neglected

(4) Implied that cask drop is the heaviest load which should be considered

References:

(1) USNRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", April 14, 1978

(2) USNRC Modifications to Reference 1, January 18, 1979

(3) USNRC Standard Review Plan, Section 3.8.4, Appendix D "Technical Position on Spent Fuel Pool Racks", Rev. O, July 1981

DSER 3.8.5 (a) and (b).

The staff requested the applicant to demonstrate that the reactor building foundation mat design complies with the requirements of the ASME Code Section III, Division 2, and also that the foundation design of Category I structures other than reactor building complies with the requirements of the ACI-349 Code as amended by Regulatory Guide 1.142. In both cases, the applicant has previously used ACI-318 Code in design. The applicant committed to perform an evaluation and submit the results by November 13, 1981 for staff review and acceptance.

The following conclusion is subject to the resolution of the above unresolved items.

The criteria that were used in the analysis, design, and construction of all the plant Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria, the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criteria 2 and 4.

Open SER Issue

3.8.5(a) Reactor Building Foundation Compliance with ASME Code

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

3.8.5(b) Category I Foundations Compliance with ACI-349/RG1.142

A response to this issue was submitted December 14, 1981, by letter number G02-81-518.

DSER 3.9.1.a (MEB-10)

3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, supports for ventilation ducting and cable trays, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary results, preoperational testing, and inservice testing of pumps and valves. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review performed under Standard Review Plan Section 3.9.1 pertains to the design transients, computer programs, experimental stress analyses and elastic-plastic analysis methods that were used in the analysis of seismic Category I ASME Code and non-Code items.

The following discusses several open issues in our review and concludes with our findings which are contingent upon resolution of these open issues.

a. In general, the transient conditions were reviewed and appear to be lacking with respect to the seismic transients. No seismic transients are specified for the majority of the components and components for which they are specified require only one OBE cycle. SRP 3.7.3 specifies that a minimum of 5 OBEs should be assumed.

b. Paragraph 3.9.1.1 (Page 3.9-1) states, "The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4." The title of

QUESTION NO. 10
(3.9.1)

No seismic transients are specified for the majority of the components and the components for which they are specified require only one OBE cycle. Justification is required.

RESPONSE

See the text revision attached.

Summation - This item is closed.

PCY:rf/45E4
8/18/81

(6)

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Code Class 1, control rod drive components, reactor assembly including core supports and reactor internals, main steam and recirculation systems. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Boiler and Pressure Vessel Code if applicable. The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4.

3.9.1.1.1 Control Rod Drive (CRD) Transients

The normal and test service load cycles used for design purposes for the 40 year life of the control rod drives are as follows:

	Transient	Category	Cycles
a.	Reactor startup/ shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure	normal/upset	10
d.	Scram test plus startup scrams	normal/upset	300
e.	Operational scrams	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Shim/drive cycles	normal/upset	1000

(10)

In addition to the above cycles, the following have been considered in the design of the CRD.

	Transient	Category	Cycles
h.	Scram with inoperative buffer	normal/upset	10
i.	Scram with stuck control blade faulted	normal/upset	1
j	OBE *	normal/upset	10
k	SSE **	faulted	1

All ASME Class I components of the CRD have been analyzed according to ASME Section III Boiler and Pressure Vessel Code.

The capability of the CRD's to withstand emergency and faulted conditions is verified by test rather than analysis.

3.9.1.1.2 CRD Housing and Incore Housing Transients

The number of transients, their cycles, and classification as considered in the design and fatigue analysis of the CRD housing and incore housing are as follows:

	Transient	Category	Cycles
a.	Normal startup & shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure tests	normal/upset	10
d.	Interruption of feedwater flow	normal/upset	80
e.	Scram	normal/upset	200

* The frequency of occurrence of this transient would indicate emergency category. However, for conservatism, the OBE condition was analyzed as an upset condition. Ten peak OBE cycles are postulated.

** SSE is a faulted condition; however in the stress analysis it was treated as emergency with lower stress limits.

(10)

	Transient	Category	Cycles
f.	~~OBE Δ~~	normal/upset	~~1~~ 10
g.	SSE~~**~~	emergency	1
CRD Housing Only			
h.	Stuck Rod Scram	normal/upset	1
i.	Scram no Buffer	normal/upset	~~1~~ 10

~~** SSE is a faulted condition; however, in the stress analysis report it was treated as emergency with lower stress limits.~~

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

10

3.9.1.1.3 Hydraulic Control Unit Transients

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

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

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

10

3.9.1.1.4 Core Support and Reactor Internals Transients

The normal and test service load cycles used for the design and fatigue analysis for the 40 year life of the CS and Internals are ~~as follows:~~ shown on Table 3.9-1.

	Transients	Category	Cycles
a.	Startup	normal/upset	120
b.	Power cycles	normal/upset	12,400
c.	Loss of feedwater heater	normal/upset	80
d.	Scram	normal/upset	198
e.	Reduction to 0% power, hot standby shutdown, & vessel flooding	normal/upset	111
f.	Unbolting	normal/upset	123
g.	Scram (Auto. blowdown & reactor overpressure)	normal/upset	2
h.	Improper start of cold recirc. loop	emergency	1
i.	Sudden start of pump in cold recirc. loop	emergency	1
j.	Improper startup	emergency	1
k.	Pipe rupture & blowdown	faulted	1

3.9.1.1.5 Main Steam System Transients

The following transients are considered in the stress analysis of the main steam piping:

Main Steam Transients

	Transient	Category	Cycles
a.	Startup	normal	121
b.	Loss of F.W. pumps isolation valves closed	upset	10
c.	Scram	upset	180
d.	Shutdown	normal	111
e.	Reactor overpressure delayed scram	emergency	1
f.	Single S/RV blowdown	upset	8
g.	Automatic blowdown	emergency	1
h.	Hydrotest	test	130
i.	OBE	upset	50

3.9.1.1.6 Recirculation System Transients

The following transients are considered in the stress analysis of the recirculation piping:

Recirculation Transients

	Transient	Category	Cycles
a.	Startup	normal	121
b.	Turbine roll and increase to power	normal	120

(10)

	Transient	Category	Cycles
c.	Loss of feedwater heater	upset	10
d.	Partial feedwater heater bypass	upset	70
e.	Scrams	upset	180
f.	Shutdown	normal	111
g.	Loss of F.W. pumps isolation valves closed	upset	10
h.	Reactor overpressure with delayed scram	emergency	1
i.	Single S/RV blow-down	upset	8
j.	Automatic blowdown	emergency	1
k.	Hydrotest	test	130
l	OBE	upset	50

3.9.1.1.7 Reactor Assembly Transients

The reactor assembly includes the reactor pressure vessel, support skirt, shroud support, and shroud plate. The cycles listed in Table 3.9-1 were specified in the reactor assembly design and fatigue analysis.

3.9.1.1.8 Main Steam Isolation Valve Transients

The main steam isolation valves are designed for the following service conditions and thermal cycles:

	Transient	Category	Cycles
a.	Pre-op @100°F/hr	normal/upset	150
b.	Startup (heating 100°F/hr)	normal/upset	120.

(16)

TABLE 3.9-1

PLANT EVENTS

		No. of Cycles
Normal, Upset, and Testing Conditions		
a.	Bolt Up*/Unbolt	123
b.	Design Hydrostatic Test	130
c.	Startup (100°F/hr Heatup Rate)**	120
d.	Daily Reduction to 75% Power*	10,000
e.	Weekly Reduction to 50% Power*	2,000
f.	Control Rod Pattern Change*	400
g.	Loss of Feedwater Heaters (80 Cycles Total):	80
h.	Operating Base Earthquake Event at Rated Operating Conditions	10/50 *****
i.	Scram:	
	1) Turbine Generator Trip, Feedwater on, Isolation Valves Stay Open	40
	2) Other Scrams	140
	3) Loss of Feedwater Pumps, Isolation Valves Closed	10
	4) Single Safety or Relief Valve Blowdown	8
j.	Reduction to 0% Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate)**	111
k.	HPCS Operation (10), SLC Operation (10)	20

10

TABLE 3.9-1 (Continued)

		No. of Cycles
Emergency Conditions		
1.	Scram:	
	1) Reactor Overpressure with Delayed Scram, Feedwater Stays on, Isolation Valves Stay Open	1***
	2) Automatic Blowdown	1***
m.	Improper Start of Cold Recirculation Loop	1***
n.	Sudden Start of Pump in Cold Recirculation Loop	1***
o.	Improper Startup with Reactor Drain Shut Off Followed by Turbine Roll and Increase to Rated Power	1***
Faulted Condition		
p.	Pipe Rupture and Blowdown	1***
q.	Safe Shutdown Earthquake at Rated Operating Conditions	1***
ASME Hydrostatic Test		
r.	1.25 x Design Pressure Hydrostatic Test (per NB 6222 and NB 3114)	10

*Applies to reactor pressure vessel only.

**Bulk average vessel coolant temperature change in any 1-hour period.

***The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

****Includes 10 maximum load cycles per event

**** 50 peak OBE cycles for NSSS piping, 10 peak OBE cycles for other NSSS equipment and components

50 peak OBE cycles are postulated for all BOP piping and components.

DSER 3.9.1.b (MEB-11)

3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, supports for ventilation ducting and cable trays, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary results, preoperational testing, and inservice testing of pumps and valves. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review performed under Standard Review Plan Section 3.9.1 pertains to the design transients, computer programs, experimental stress analyses and elastic-plastic analysis methods that were used in the analysis of seismic Category I ASME Code and non-Code items.

The following discusses several open issues in our review and concludes with our findings which are contingent upon resolution of these open issues.

a. In general, the transient conditions were reviewed and appear to be lacking with respect to the seismic transients. No seismic transients are specified for the majority of the components and components for which they are specified require only one OBE cycle. SRP 3.7.3 specifies that a minimum of 5 OBEs should be assumed.

b. Paragraph 3.9.1.1 (Page 3.9-1) states, "The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4." The title of

Table 3.7-4 is "Reactor Building-Seismic Analysis Natural Frequency and Natural Period." Reference to this Table appears to be in error. Clarification is requested.

c. Paragraph 3.9.1.1.13 (Page 3.9-14) states that the applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15. This Table has not been completed and therefore remains an open item.

d. Computer programs were used in the analysis of specific components. A list of the computer programs that were used in the dynamic and static analyses to determine the structural and functional integrity of these components is included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs be included. While the required verification is provided for most computer programs, it is lacking for several. The applicant must provide methods of verification for all of the listed computer programs.

e. The computer code utilized in the analysis of the ECCS Pump Motor Rotor Shafts addressed in paragraphs 3.9.1.2.4, ECCS Pumps and Motors, is not identified. This code should be identified and data presented for the validity and applicability for use of this code.

f. The Orificed Fuel Support experimental stress analysis discussed in Paragraph 3.9.1.4.2.5 (Pages 3.9-19 and 20) is not adequate to establish the validity of this program. Additional details concerning this test program are required. In addition, it states that the allowable stress limits were arrived at by applying a 0.65 quality factor to the ASME Code allowables of 1.5 S_m for upset. The basis for the 0.65 factor is required.

g. The statement is made in Paragraph 3.9.1.4.1.2, Hydraulic Control Unit, that "These stresses were obtained by assuming that two HCUs were braced

Table 3.7-4 is "Reactor Building-Seismic Analysis Natural Frequency and Natural Period." Reference to this Table appears to be in error. Clarification is requested.

c. Paragraph 3.9.1.1.13 (Page 3.9-14) states that the applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15. This Table has not been completed and therefore remains an open item.

d. Computer programs were used in the analysis of specific components. A list of the computer programs that were used in the dynamic and static analyses to determine the structural and functional integrity of these components is included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs be included. While the required verification is provided for most computer programs, it is lacking for several. The applicant must provide methods of verification for all of the listed computer programs.

e. The computer code utilized in the analysis of the ECCS Pump Motor Rotor Shafts addressed in paragraphs 3.9.1.2.4, ECCS Pumps and Motors, is not identified. This code should be identified and data presented for the validity and applicability for use of this code.

f. The Orificed Fuel Support experimental stress analysis discussed in Paragraph 3.9.1.4.2.5 (Pages 3.9-19 and 20) is not adequate to establish the validity of this program. Additional details concerning this test program are required. In addition, it states that the allowable stress limits were arrived at by applying a 0.65 quality factor to the ASME Code allowables of 1.5 S_m for upset. The basis for the 0.65 factor is required.

g. The statement is made in Paragraph 3.9.1.4.1.2, Hydraulic Control Unit, that "These stresses were obtained by assuming that two HCUs were braced

QUESTION NO. 11
(3.9.1)

Paragraph 3.9.1.1, Design Transients, referring to Table 3.7-4, "Reactor Building-Seismic Analysis Natural Frequency and Natural Period," appears to be in error. Clarification is required.

RESPONSE

The statement is deleted. See the text revision attached.

Summation - This item is closed.

PCY:rf/45E5
8/18/81

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

3.9.1.1 Design Transients

This section shows the transients which are used in the design of the ASME Code Class 1, control rod drive components, reactor assembly including core supports and reactor internals, main steam and recirculation systems. The number of cycles or events for each transient are included. The design transients shown in this section are included in the design specifications for the components. Transients or combinations of transients are classified with respect to the component operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing" in the ASME Boiler and Pressure Vessel Code if applicable. The cycles due to SSE and OBE used in the fatigue analysis are shown in Table 3.7-4.

3.9.1.1.1 Control Rod Drive (CRD) Transients

The normal and test service load cycles used for design purposes for the 40 year life of the control rod drives are as follows:

	Transient	Category	Cycles
a.	Reactor startup/ shutdown	normal/upset	120
b.	Vessel pressure tests	normal/upset	130
c.	Vessel overpressure	normal/upset	10
d.	Scram test plus startup scrams	normal/upset	300
e.	Operational scrams	normal/upset	300
f.	Jog cycles	normal/upset	30,000
g.	Shim/drive cycles	normal/upset	1000

DSER 3.9.1.c (MEB-12) DSER 3.9.1.f (MEB-16)
DSER 3.9.1.d (MEB-13) DSER 3.9.1.g (MEB-17)
DSER 3.9.1.e (MEB-15)

Table 3.7-4 is "Reactor Building-Seismic Analysis Natural Frequency and Natural Period." Reference to this Table appears to be in error. Clarification is requested.

c. Paragraph 3.9.1.1.13 (Page 3.9-14) states that the applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15. This Table has not been completed and therefore remains an open item.

d. Computer programs were used in the analysis of specific components. A list of the computer programs that were used in the dynamic and static analyses to determine the structural and functional integrity of these components is included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs be included. While the required verification is provided for most computer programs, it is lacking for several. The applicant must provide methods of verification for all of the listed computer programs.

e. The computer code utilized in the analysis of the ECCS Pump Motor Rotor Shafts addressed in paragraphs 3.9.1.2.4, ECCS Pumps and Motors, is not identified. This code should be identified and data presented for the validity and applicability for use of this code.

f. The Orificed Fuel Support experimental stress analysis discussed in Paragraph 3.9.1.4.2.5 (Pages 3.9-19 and 20) is not adequate to establish the validity of this program. Additional details concerning this test program are required. In addition, it states that the allowable stress limits were arrived at by applying a 0.65 quality factor to the ASME Code allowables of 1.5 S_m for upset. The basis for the 0.65 factor is required.

g. The statement is made in Paragraph 3.9.1.4.1.2, Hydraulic Control Unit, that "These stresses were obtained by assuming that two HCUs were braced

together back to back ..." Are the units actually tied together as assumed? Additional details are required.

Subject to resolution of these open issues, our findings are as follows:

The methods of analysis that the applicant has employed in the design of all seismic Category I ASME Code Class 1, 2, and 3 components, component supports, reactor internals, and other non-Code items are in conformance with Standard Review Plan Section 3.9.1 and satisfy the applicable portions of General Design Criteria 2,4,14, and 15.

The criteria used in defining the applicable transients and the computer codes and analytical methods used in the analyses provide assurance that the calculations of stresses, strains, and displacements for the above noted items conform with the current state-of-the-art and are adequate for the design of these items.

3.9.2 Dynamic Testing and Analysis

The review performed under Standard Review Plan Section 3.9.2 pertains to the criteria, testing procedures, and dynamic analyses employed by the applicant to assure the structural integrity and operability of piping systems, mechanical equipment, reactor internals and their supports under vibratory loadings.

3.9.2.1 Preoperational Vibration and Dynamic Effects Piping Tests

The preoperational vibration test program will be conducted during startup and initial operation. The purpose of these tests is to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the ASME Section III Code and to confirm that no unacceptable restraint of normal thermal motion occurs. We have identified the following open issues in our review.

QUESTION NO. 12
(3.9.1)

Table 3.9-15, Applicable Seismic Cycle Loading, is indicated as "Later". Provide a schedule for its inclusion in the FSAR.

RESPONSE

Table 3.9-15 is deleted.

The statement in Section 3.9.1.1.13 that references Table 3.9-15 has also been deleted.

Summation - This item is closed.

Pressure Transient		Cycles
g.	110% design pressure at 575°F	1
h.	1300 psi at 100°F installed hydrostatic test	130
i.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.1.13 Balance of Plant Transients

The transients used in design and fatigue analysis of the balance of plant components are listed in Table 3.9-1, with an exception that 50 maximum stress cycles due to OBE are used in fatigue evaluation. ~~A complete list of applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15.~~

3.9.1.2 Computer Programs Used in Analysis

The following sections list the computer programs used in the analysis of specific components. These programs are described in 3.12.

3.9.1.2.1 Reactor Vessel

The following programs are used in the analysis of the Reactor Vessel:

a. CB&I Program 711 "GENOZZ"

b. CB&I Program 948 "NAPALM"

c. CB&I Program 1027

d. CB&I Program 846

e. CB&I Program 781 "KALNINS"

f. CB&I Program 979 "ASFAST"

g. CB&I Program 766 "TEMAPR"

h. CB&I Program 767 "PRINCESS"

TABLE 3.9-15

APPLICABLE SEISMIC CYCLIC LOADING

~~(LATER)~~

DELETED

QUESTION NO. 13
(3.9.1)

Methods of verification are required for all NSSS computer codes used in the analysis.

RESPONSE

The NSSS programs can be divided into two categories.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a) MASS
(b) SNAP (MULTISHELL)
(c) GASP
(d) NOHEAT
(e) FINITE
(f) DYSEA
(g) SHELLS
(h) HEATER
(i) FAP-71 *
(j) CREEP-PLAST
(k) PISYS
(l) ANSI7
(m) SAP4G
(n) FTFLG01
(o) ANSYS
(p) BSTIF01

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

Byron Jackson Program

RTRMEC

CB&I Programs

(a) 711 GENOZZ
(b) 948 NAPALM *
(c) 1027
(d) 846
(e) 781 KALNINS
(f) 979 ASFAST
(g) 766 TEMAPR
(h) 767 PRINCESS
(i) 928 TGRV
(j) 962 E0962A
(k) 984
(l) 992 GASP
(m) 1037 DUNHAM'S
(n) 1335
(o) 1606 & 1657 HAP
(p) 1635
(q) 953

Accordingly, the FSAR text is revised as attached.

Summation - This item is closed pending NRC audit.

* To be audited by NRC.

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Pressure Transient		Cycles
g.	110% design pressure at 575°F	1
h.	1300 psi at 100°F installed hydrostatic test	130
i.	1670 psi at 100°F installed hydrostatic test	3

3.9.1.1.13 Balance of Plant Transients

The transients used in design and fatigue analysis of the balance of plant components are listed in Table 3.9-1.

A complete list of applicable seismic cyclic loading for operating basis earthquake is shown in Table 3.9-15.

~~3.9.1.2 Computer Programs Used in Analysis~~

~~The following sections list the computer programs used in the analysis of specific components. These programs are described in 3.12.~~

~~3.9.1.2.1 Reactor Vessel~~

~~The following programs are used in the analysis of the Reactor Vessel:~~

~~a. CB&I Program 711 "GENOZZ"~~

~~b. CB&I Program 948 "NAPALM"~~

~~c. CB&I Program 1027~~

~~d. CB&I Program 846~~

~~e. CB&I Program 781 "KALNINS"~~

~~f. CB&I Program 979 "ASFAST"~~

~~g. CB&I Program 766 "TEMAPR"~~

~~h. CB&I Program 767 "PRINCESS"~~

3.9.1.2 Computer Programs Used in Analysis

The following sections discuss computer programs used in the analysis of the major safety-related components. (Computer programs were not used in all components, hence not all components are listed.) The NSSS programs can be divided into two categories.

GE Programs

The verification of the following GE programs has been performed in accordance with the requirements of 10CFR50, Appendix B. Evidence of the verification of input, output, and methodology is documented in GE Design Record Files.

(a) MASS
(b) SNAP (MULTISHELL)
(c) GASP
(d) NOHEAT
(e) FINITE
(f) DYSEA
(g) SHELL5
(h) HEATER
(i) FAP-71
(j) CREEP-PLAST
(k) PISYS
(l) ANSI7
(m) SAP4G
(n) FTFLG01
(o) ANSYS
(p) BSTIF01

Vendor Programs

The verification of the following two groups of vendor programs is assured by contractual requirements between GE and the vendors. Per the requirements, the quality assurance procedure of these proprietary programs used in the design of N-stamped equipment is in full compliance with 10CFR50, Appendix B.

Byron Jackson Program

RTRMEC

CB&I Programs

(a) 711 GENOZZ
(b) 948 NAPALM
(c) 1027
(d) 846
(e) 781 KALNINS
(f) 979 ASFAST
(g) 766 TEMAPR
(h) 767 PRINCESS
(i) 928 TGRV
(j) 962 E0962A
(k) 984
(l) 992 GASP
(m) 1037 DUNHAM'S
(n) 1335
(o) 1606 & 1657 HAP
(p) 1635
(q) 953

3.9.1.2.1 Reactor Vessel and Internals

3.9.1.2.1.1 Reactor Vessel

CB&I Programs (a) through (q) listed above are used to analyze the reactor pressure vessel. Detailed descriptions are provided in Section 3.12.

3.9.1.2.1.2 Reactor Internals

The following computer programs are used in the analysis of the core support structures and other safety-related reactor internals: MASS, SNAP (MULTISHELL) GASP, NOHEAT, FINITE, DYSEA, SHELL5, HEATER, FAP-71, and CREEP-PLAST. Detailed descriptions of these programs are provided in Section 4.1.

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i. CB&I Program 928 "TGRV"

j. CB&I Program 962 "E0962A"

k. CB&I Program 984

l. CB&I Program 992 "GASP"

m. CB&I Program 1037 "DUNHAM'S"

n. CB&I Program 1335

o. CB&I Programs 1606 and 1657 "HAP"

p. CB&I Program 1635

q. CB&I Program 953

3.9.1.2.2 Piping

The following programs are used in the analysis of piping:

a. ADLPIPE

b. DYNAMIC ANALYSIS OF PIPING SYSTEMS

c. PLATE PANEL, SPACE STRUCTURAL ANALYSIS (MASS)

d. SHELL ANALYSIS PROGRAM (MULTISHELL)

e. TIME DEPENDENT PIPE FORCE

f. PIPE DYNAMIC ANALYSIS PROGRAM (PDA)

3.9.1.2.3 Recirculation Pump

No computer programs were used in the design of the recirculation pumps.

3.9.1.2.4 ECCS Pumps and Motors

An equivalent static computer analysis was performed on the ECCS pump motor rotor shafts. The model consisted of lumped masses simulating the distribution of mass in the system,

3.9.1.2.2 Piping

3.9.1.2.2.1 Piping Analysis Program/PISYS

PISYS is a computer code specialized for piping load calculations. It utilizes selected stiffness matrices representing standard piping components, which are assembled to form a finite element model of a piping system. The technique relies on dividing the pipe model into several discrete substructures, called pipe elements, which are connected to each other via nodes called pipe joints. It is through these joints that the model interacts with the environment, and loading of the structure becomes possible. PISYS is based on the linear classical elasticity in which the resultant deformation and stresses are proportional to the loading, and the superposition of loading is valid.

PISYS has a full range of static and dynamic analysis options which include distributed weight, thermal expansion, differential support motion modal extraction, response spectra, and time history analysis by modal or direct integration. The PISYS program has been benchmarked against five Nuclear Regulatory Commission piping models for the option-of-response-spectrum analysis and the results are documented in a report to the Commission, "PISYS Analysis of NRC Problem," NEDO-24210, August, 1979.

3.9.1.2.2.2 Component Analysis/ANSI7

The ANSI 7 computer program determines stress and accumulative usage factors in accordance with NB-3600 of the ASME Code, Section III. The program was written to perform stress analysis in accordance with the ASME Code sample problem, and has been verified by reproducing the results of the sample problem analysis.

3.9.1.2.3 ECCS Pumps and Motors

3.9.1.2.3.1 Rotor Assembly Analysis Program/RTRMEC

RTRMEC is a computer program which calculates and displays results of mechanical analysis of motor rotor assembly when acted upon by external forces at any point along shaft (rotating parts only). The shaft deflection due to magnetic and centrifugal forces was analyzed. The calculation for the seismic condition assumes that the motor is operating and that the seismic, magnetic, and centrifugal forces all act simultaneously and in phase on the rotor-shaft assembly. Note that the distributed rotor assembly weight is lumped at the various stations, with the shaft weight at a station being the sum of one-half the weight of the incremental shaft length just before the station, plus one-half the weight of the adjacent incremental shaft length just after the station. Bending and shear effects are accounted for in the calculations.

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3.9.1.2.3.2 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the ECCS pump/motor systems. This is a general structural analysis program for static and dynamic analysis of linear elastic complex structures. The finite element displacement method is used to solve the displacements and stresses of each element of the structure. The structure can be composed of unlimited number of three-dimensional truss, beam, plate, shell, solid, plate strain-plane stress and spring elements that are axisymmetric. The program can treat thermal and various forms of mechanical loading. The dynamic analysis includes mode superposition, time history, and response spectrum analysis. Seismic loading and time-dependent pressure can be treated. The program is versatile and efficient in analyzing large and complex structural systems. The output contains displacements of each nodal point as well as stresses at the surface of each element.

3.9.1.2.3.3 Effects of Flange Joint Connections/FTFLG01

The flange joints connecting the pump bowl castings are analyzed using FTFLG01. This program uses the local forces and moments determined by SAP4G to perform flat flange calculations in accordance with the rules set forth in Appendix II and Section III of the ASME Boiler and Pressure Vessel Code.

3.9.1.2.3.4 Structural Analysis of Discharge Head/ANSYS

ANSYS is used to analyze the pump discharge head flange and bolting taking into account of the prying action developed by the flat face contact surface. The program is described in detail in 3.12.

3.9.1.2.4 RHR Heat Exchangers

3.9.1.2.4.1 Structural Analysis Program/SAP4G

SAP4G is used to analyze the structural and functional integrity of the RHR heat exchangers. The description of this program is provided in Subsection 3.9.1.2.3.2.

3.9.1.2.4.2 Local Stiffness Calculations/BSTIF01

BSTIF01 is used to estimate the local stiffness of the heat exchanger shell at the attachment point of the supports. The method used in this program is based on the shell stiffness calculations by P. P. Bijlaard as groundwork for Welding Research Council Bulletin 107. The results of BSTIF01 are used to determine equivalent beam properties of the lower and upper heat exchanger support bracket to shell attachments included in the finite element model of the heat exchanger.

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~~connected by massless elastic members, simulating the distribution of shaft stiffness. The analysis was performed iteratively to obtain compatibility between the rotor displacements and the magnetic and centrifugal forces acting on the rotor.~~

~~All other analysis of specific motor components and pump components consisted of hand calculations.~~

~~3.9.1.2.5 RHR Heat Exchangers~~

~~Following are the computer programs used in dynamic and static analysis to determine structural and functional integrity of the RHR heat exchangers:~~

~~Support Load Seismic Analysis (ED-6)~~

~~Stress Analysis of Supports (ED-8)~~

~~3.9.1.2.6 Other Computer Programs Used in Analyses~~

~~Other computer programs used in the dynamic and static analyses of structural and functional integrity of Seismic Category I systems, components, equipment and supports are listed in 4.1.4 and 3.12~~

3.9.1.3 Experimental Stress Analysis

When experimental stress analysis is used in lieu of analytical methods for Seismic Category I ASME Code items, the requirements for experimental testing enumerated in the ASME Code which are applicable for the specific components under test shall be applied. When testing is required for Seismic Category I non-ASME Code parts account shall be taken of size effects and dimensional tolerances which exist between the actual part and the test part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts, to assure that the loads obtained from the test are a realistic or conservative representation of the load carrying capability of the actual structure under the postulated loading.

Results of both ISOFINITE and NASTRAN are given in Table 3.12-2. As can be seen, there is close correlation between the deflections, with NASTRAN giving larger values throughout the flued head than ISOFINITE. This is due to the lack of rotational freedom at nodes with NASTRAN over the more flexible shear elements in ISOFINITE. This leads to prediction of higher stresses using ISOFINITE (as can be seen by comparing pages 2 and 3 of Table 3.12-2). The computer program ISOFINITE is therefore a conservative method for determining stresses in flued head fittings.

This program is referred to in 3.8.6.4.4.

3.12.10 ADLPIPE

ADLPIPE is a digital computer program developed by the Arthur D. Little Co. and used for static and dynamic analyses of complex piping systems. Input data preparation uses piping language and output information is presented for easy interpretation. The input data may be pre-processed and plots made for input and model evaluation. To aid in rapid data preparation there are many input error diagnostics. The output automatically includes a stress analysis as required by ANSI B31.1 (1967); B31.1 (1973); ASME Code Section III, Class 1, Class 2 and Class 3 (1971 and 1974). The ASME Code, Section III, Class 1 analysis includes calculation of fatigue usage factor and simplified elastic-plastic analysis. All forces, moments, deflections, rotations and a summary stress report are included in the output. Additionally, the program has orthographic, isometric, and stereoscopic plotting capability to aid checking input and interpreting computed results.

The piping system is modeled as a series of sections that lie between network points. A section is composed of straight and curved members, and each member may have common or different loads and physical properties. The network points may be free, partially or fully restrained and have specified displacements that represent thermal anchor displacements or seismic anchor motion. Intermediate springs to ground or joining other members may be placed within the section to represent spring hangers, pipe bellows, skew and guided restraints, support and equipment stiffness. Transfer matrix techniques are used to reduce the size of the stiffness matrix.

3.12.10 ANSYS

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

1. Structural analysis including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

2. One-dimensional fluid flow analyses.

3. Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

4. An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

5. Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures.

6. Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

The program is maintained current by Swanson Analysis Systems, Inc. of Pittsburgh, Pennsylvania and is supplied to General Electric for use on the Honeywell 6000.

The ANSYS program has been used for productive analyses since early 1970. Users now include the nuclear, pressure vessels and piping, mining, structures, bridge, chemical, and automotive industries, as well as many consulting firms.

The static loads on the piping system may be thermal, deadweight, static "g" seismic loads, externally applied forces and moments, and wind loads. The dynamic loads are computed using normal mode theory and seismic response spectra or time history forcing functions in one or more directions.

The approach used in ADLPIPE to compute the response of piping system to ground shock inputs is based upon a normal mode or modal superposition method. The formulation in terms of normal modes, which is particularly advantageous for transient reponse problems, follows generally the form discussed by Young (Reference 3.12-15).

The first step in the application of this method is the determination of the natural frequencies and mode shapes of the free vibrations of the system. For a conservative linearly-elastic lumped mass system, the governing matrix equation is:

$$M u + K_R u = 0$$

where

M is the (diagonal) inertia matrix,

K_R is the stiffness matrix, and

u is the column matrix of the displacement coordinates.

The stiffness matrix K_R utilized in the dynamics formulation differs from the stiffness matrix K_R developed by ADLPIPE for the network points. The latter matrix, developed by transfer matrix techniques, includes mass points and interior network points. The stiffness matrix for the dynamics formulation requires stiffness values at mass points only. Thus, K_R is a reduced form of K_R, and can be shown to be equal to:

$$K_R = A - B' \cdot E^{-1} \cdot D$$

where:

$$K = \begin{matrix} A & B \\ D & E \end{matrix}$$

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and A = Mass points sub-matrix

B,D = Coupling sub-matrices

E = Branch points submatrix.

Determination of the natural modes can proceed by any one of several methods. The two eigenvalue routines used in ADLPIPE are the Jacobi (Reference 3.12-16) rotation scheme and the Givens-Householder (Reference 3.12-16) scheme; the latter has been modified to incorporate a suggestion made by Wilkinson (Reference 3.12-17). In the Jacobi routine, the operations are carried out in core memory and the number of degrees of freedom is limited by available core. The Givens-Householder routine is unlimited by core utilization of secondary storage and produces the lowest eigenvalues and associated eigenvectors.

For a system having N degrees of freedom, the eigenvalue routines will produce up to N eigenvalues (natural frequencies) ωi (i=1 ...N) and up to N sets of eigenvectors, ϕij (i=1, ...N, j=1, 2, ...N). The j^{th} column of the (N x N) array ϕij is called the eigenvector for the j^{th} mode while the array itself is called the modal matrix.

The normal mode formulation of the response of a lumped system to a shock displacement can be carried out by considering the kinetic and potential energies of the loaded system. Assuming that the elastic displacement of the i^{th} coordinate is u_i, and the total displacement equal to $u_i + s_i$, where s_i is the shock displacement of the i^{th} coordinate "normal" coordinates q_n (t) and $p^{(t)}$ (t) are defined by the linear transformation:

$$u_i(t) = \sum_{n=1}^{N} \phi_{in} q_n(t)$$

$$s_i(t) = \sum_{n=1}^{N} \phi_{in} p_n(t)$$

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where ϕ_{in} is the i^{th} element of the n^{th} eigenvector. These transformations are useful because the resulting equations of motion in terms of the normal coordinates are completely decoupled from one another.

The kinetic energy T and the potential energy V of the system, in terms of the normal coordinates, are given by (reference 3.12-15):

$$T = \frac{1}{2} \sum_{i=1}^{N} m_i \dot{q}_i^2 + s \sum_{i=1}^{N} m_i \sum_{n=1}^{N} \phi_{in} \dot{q}_n + \frac{1}{2} \dot{s}^2 \sum_{i=1}^{N} m$$

where: m_i are the individual mass elements, and

M_i = generalized mass for the i^{th} mode, defined by the relation:

$$M_i = \sum_{j=i}^{N} m_j \phi_{ji}^2$$

Substitution of these energy expressions into Lagrange's equation leads to the equations of motion:

$$\ddot{q}_n + \omega_n^2 q_n = -\bar{p}_n$$

The solution of these equations for the modal amplitudes q_n is:

$$q_n(t) = \frac{1}{\omega_n} \int_0^t \bar{p}_n(T) \sin\omega_n (t-T) dT$$

where T is a dummy variable of integration.

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From the transformation equation, we have

$$\bar{p}_n = \sum_{\ell} \left[\phi^{-1}_{n\ell}\bar{s}_{\ell}\right]$$

where $\Sigma\phi^{-1}_{n\ell}\phi_{\ell m} = I_{nm}$ = the identity matrix.

Hence,

$$q_n(t) = \frac{1}{\omega_n}\left[\sum_{\ell}\left(\phi^{-1}_{n\ell}\right) \int_0^t \bar{s}_{\ell}(T)\sin\omega_n(t-T)\,dT\right]$$

Defining $R_n(t) = \frac{1}{\omega_n} \int_0^t \bar{s}_{\ell}(T)\sin\omega_n(t-T)\,dt$

the modal response can be written simply as:

$$q_n(t) = \left(\sum_{\ell}\phi^{-1}_{n\ell}\right)\left(R_n(t)\right)$$

The expression $\sum_{\ell}\phi^{-1}_{n\ell}$ represents the portion of the maximum modal response developed by each normal coordinate, and may be thought of as a measure of the extent of which the n^{th} normal mode participates in the synthesis of the total response of the structural system. As such, the square array $\phi^{-1}_{n\ell}$, which is the inverse of the modal matrix, is termed the modal participation matrix, with each element of the matrix corresponding to the "participation" in the overall response synthesis of mode n, and mass point ℓ.

The term $R_n(t)$, expressed by the convolution integral, represents the response of mode n as a function of time, assuming that mode n is uncoupled from the other modes of the system; i.e., $R_n(t)$ is the response of a single degree of freedom system to the transient loading given by s(t).

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The three dimensional shock input displacement, D_{ln}, is given in terms of a maximum-valued spectrum (such as the Housner earthquake input spectra) for each principal axis. For instance, the vertical response may be different from the horizontal response. Thus, the prescribed input for each mode is the maximum value of the response $R_n(t)$ developed during the overall duration of the response. (These values are obtained, for example, by measurement with displacement meters such as cantilever reed gages which record the peak value of the displacement during the response period.)

Thus,

$$(D_l)_n = \left| R_n(t) \right| \text{maximum}$$

Therefore, the modal amplitudes become

$$q_n(t) \leq \sum_l \left[\phi^{-1}_{nl}\right] (D_l)_n$$

For each normal mode, the amplitudes at each coordinate i are given by

$$x_i = \phi_{in} q_n$$

where:

$$u_i = \sum_{n=i}^{N} x_i$$ = elastic amplitude at coordinate i summed over all modes.

This then provides a set of displacements, x_i, for each of the n modes. These individual sets of displacements can then be applied to the system as equivalent static deflections on a mode-by-mode basis. The corresponding network forces are obtained by ADLPIPE.

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

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

~~This program is referred to in 3.9.1.2.2.~~

3.12.11 RELAP3

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

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

This program is referred to in 3.6.2.2.1b and 3.6.2.3.1.

3.12.11.1 RELAP4/MOD5

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

QUESTION NO. 15
(3.9.1)

The computer code utilized in the analysis of the ECCS Pump Motor Rotor Shafts addressed in Paragraph 3.9.1.2.4, ECCS Pumps and Motors, is not identified. This code should be identified and data presented for the validity and applicability for use of the code.

RESPONSE

Referring to the response to Question No. 13, RTRMEC was used by the motor vendor to estimate the ECCS motor shaft displacement. The results of the calculation have been verified by (1) comparison with motor rotor bend test data and (2) comparison with the SAP4G results obtained by GE. The comparison demonstrated the conservatism of RTRMEC.

Summation - This item is closed.

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QUESTION NO. 16
(3.9.1)

Provide additional details concerning the test program performed on the orificed fuel support to establish the validity of the program. In addition, provide justification for using the allowable limits by applying a 0.65 quality factor to the ASME Code allowables of 1.5 Sm for upset condition.

RESPONSE

1. Test Program

The following is a detailed description of the test program. (Note: WNP-2's orificed fuel support is not required to conform with the ASME Code; however, the test program is designed to conform to the code in order to verify the design adequacy.)

Two separate tests were conducted, ~~and~~ the each test was designed to be in conformance with Appendix II of th ASME code Section III. The first test series verified the structural capability of the fuel support casting to sustain vertical design loads. A production fuel support was stresscoated and subjected to an extremely high vertical load to identify the location and principal stress directions of the highest stressed regions. A second fuel support was instrumented with strain gauges: 12 uniaxial gauges were used where the principal stress directions were known from the previous stresscoat test. Six rosettes were used where the principal stress axes could not readily be determined. (All the gauges used in the experimental stress analysis were put in the regions of highest stress as determined by the previous stresscoat test.) The fuel support was mounted in a fixture simulating the geometric characteristics of both the load and support in the reactor. Vertical loads only were applied, simulating the weight load of the fuel assemblies.

It was found that the fuel support could sustain a vertical load of 104,000 pounds before the onset of yielding in the highest stressed region. This 104,000 pound load represents a safety factor in excess of 35 based on yielding over the normal applied vertical load.

A second series of tests were conducted to investigate the resulting stresses induced in the fuel support by a horizontal (or lateral) load applied by the fuel assemblies during a seismic event. A fuel support was instrumented with 15 three-element rosette strain gauges. The location of these gauges were determined from an initial computer analysis, and represented the areas of highest stress plus a few key locations of minimal material thickness.

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The test fixtures used were designed to apply equal loads on all four pods. This was achieved by using two hydraulic cylinders to load two spreader bars. The load was transmitted into each spreader bar through balls which prevented moment build-up. Each spreader bar then loaded two arms, which in turn loaded dummy fuel lower tie plates. At the interface of the tie plates in the fuel support, the dimensions of these dummy tie plates were identical to those used in the production components. During loading, weight was placed at the top of the load arms approximately in the center of the fuel support. This loading simulated a vertical load which would be present due to the fuel assembly weight.

During the initial phases of the testing, it was discovered that the stresses induced by a horizontal load were a maximum when the applied vertical load was a minimum. Because the fuel support is not attached to the guide tube and sits on a chamfered seat on the guide tube provided for that purpose, it was found that an increased downward vertical load actually enhanced the fuel support's ability to sustain a horizontal load. (With increased vertical load, additional rigidity was provided to the fuel support casting by the guide tube.)

A load cell was calibrated and installed on the lower hydraulic cylinder. Load data was recorded on a continuous recorder, and strain gauge data was recorded on a multi-channel recorder. The total applied load was twice the load cell readings.

The first horizontal loading applied simulated the ASME code upset condition. For this condition the total vertical load was calculated to be just under 1,000 pounds with a horizontal load of 2,600 pounds being applied. The calculated vertical load applied to the fuel casting included its weight, the upward component of a 1/2g seismic load, and the differential pressure across the fuel and the fuel support. The 2,600 pound load was taken from the fuel support design specification for the upset event. A horizontal test load of 3,000 pounds was applied to compensate for possible increased hydraulic piston friction, changes in friction due to a small amount of misalignment and/or cocking of the load arm in relation to the piston travel direction.

The test results simulating the upset horizontal loading conditions produced a maximum stress of 10,833 psi. The differential pressure stresses across the castings were computed. The 1,580 psi value obtained from the computation was then added to the test results. (Differential pressures across the fuel support were not simulated in the test program.) The total resultant stress was 12,413 psi for the upset condition. The total stress resultant was less than the ASME code allowable of 15,580 psi for the upset condition.

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9/23/81

A second series of test loadings were applied to the support casting and were designed to simulate the faulted conditions. No vertical load was applied during this phase of the testing because of the net result of 1g downward force due to gravity and the 1g upward component of force due to the safe shutdown seismic faulted event. The horizontal test load was applied to simulate 5,200 pounds of force for the faulted event.

Testing simulating the faulted horizontal loading produced a maximum stress intensity of 21,225 psi. A computed stress value of 1,580 psi for the internal pressure was added to the test result similar to that of the upset event described above. The addition of these two stresses resulted in a maximum stress intensity of 23,505 psi, which is significantly less than the 35,400 psi allowed by ASME code for the faulted conditions.

2. Quality Factor

The 0.65 Quality factor accounts for the fact that not all castings are fully volumetrically examined. It is specified in the ASME Code, 1974 Edition, Summer 1975 Addenda, Paragraph NG-2571.2(a).

Summation - this item is closed.

NOTE: Response on "Quality Factor" has been revised to cite correct code edition.

Question 17

Expand Paragraph 3.9.1.4.1.2 (page 3.9-18) to describe the actual mounting of the hydraulic control units and to justify the validity of the assumption utilized in the FSAR.

Response

Please refer to revised 3.9.1.4.1.2 (page 3.9-18) for the information requested.

Summation - This item is closed.

No analysis has been made for the non-code components of the CRD for the abnormal condition.

The design adequacy of non-code components of the CRD has been verified by extensive testing programs on components parts, specially instrumented prototype drives and production drives. The testing included postulated abnormal events as well as the service life cycle listed in 3.9.1.1.1.

3.9.1.4.1.2 Hydraulic Control Unit

The Hydraulic Control Unit (HCU) was analyzed for the SSE faulted conditions, through the implementation of the computer code SAMIS (See 3.12). Using the method of "Sum of Absolute Values of the Modal Loads," the maximum stress on the HCU frame was calculated to be 54,310 psi. The maximum allowable for SSE is 60,000 psi for the HCU. These stresses were obtained by ~~assuming that~~ tying two HCUs ~~are braced~~ together back to back ~~on the "I" beam~~ at the top, mid-depth and bottom of the HCUs; add attached INSERT (A)

The fundamental frequency of the HCU is close to the frequency at which peak seismic shock will occur. This results in overstressed conditions in the piping connected to the HCU during the safe shutdown earthquake (SSE). By the application of bolt-on stiffening struts to the HCU frames and diagonal braces along the rows of installed HCU's, the fundamental frequency is raised sufficiently to avoid peak seismic response. The stresses in the connected piping are thus reduced to acceptable values.

The analysis of the HCU under faulted condition loads establishes the structural integrity of the system.

3.9.1.4.1.3 CRD Housing

The SSE is classified as a faulted condition; however, in the CRD housing analysis the SSE event has been treated as an emergency condition. The maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing. The stresses are within elastic limits and are shown in Table 3.9-2(v).

Response to MEB SER
Question #17

Insert A to FSAR Page 3.9-18

each pair of tied HCUs is supported in each of three mutually perpendicular directions by means of struts and diagonal bracing connected from the HCUs to a three dimensional seismic support frame enclosing rows of HCUs and anchored to the concrete foundation. See attached Figure Q 17-1.

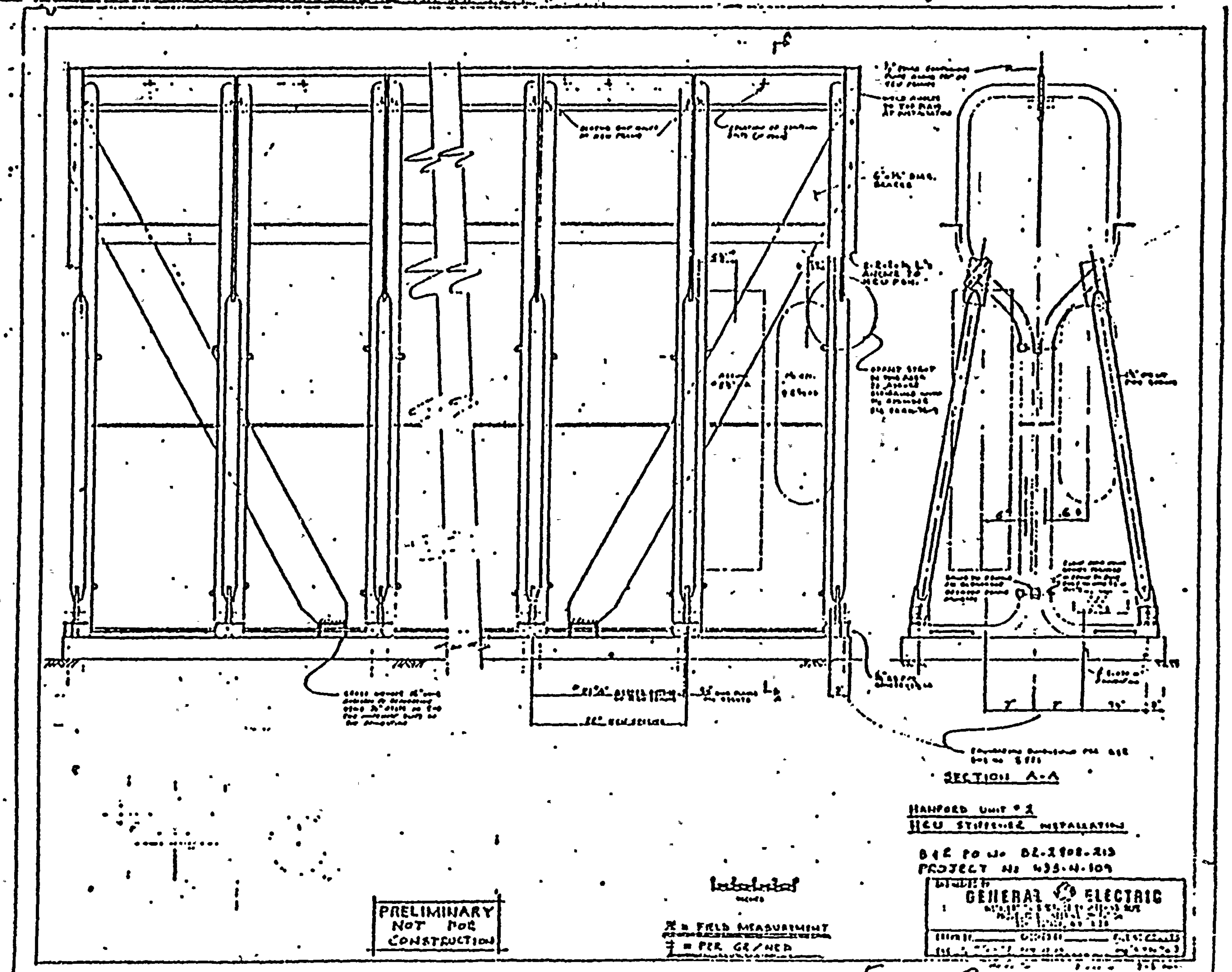
SECTION A-A
HANFORD UNIT # 2
HCU STIFFENER INSTALLATION
B&R PO NO. 02-2908-213
PROJECT NO. 435-N-109
GENERAL ELECTRIC
PRELIMINARY
NOT FOR
CONSTRUCTION
* = FIELD MEASUREMENT

FIG. B. 17-1

DSER 3.9.2.1.a (MEB-18) DSER 3.9.2.1.c (MEB-20)
DSER 3.9.2.1.b (MEB-19) DSER 3.9.2.1.d (MEB-21)

a: The applicant should provide a commitment in the FSAR stating that all required piping restraints, components and component supports have been installed in the piping system prior to testing.

b. The applicant's proposed preoperational test program covers the vibration and dynamic effects. However, the thermal expansion effects required in SRP 3.9.2.II-1.d, e, and f are not adequately addressed. The thermal motion monitoring program should deal specifically with verification of snubber movement, adequate clearances and gaps to allow free movement of the pipe during heat-up and cooldown and should include acceptance criteria and test procedure. Additional information on this program is required.

c: The applicant has not given a clear description of the acceptance criteria for steady-state piping vibrations. The staff's position is that acceptance limits for vibration should be based on half the endurance limit as defined by the ASME Code at 10^6 cycles.

d. Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety-related systems and components, it is requested that the operability program for snubbers should be included and documented by the preservice inspection and preoperational test program. We will require the applicant's response to the letter from R. Tedesco to R. Ferguson, "Preservice Inspection and Testing of Snubbers," dated March 6, 1981.

Subject to resolution of these open issues, our findings will be as follows:

The vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high and moderate energy piping, and all associated systems, restraints and supports is an acceptable program. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist

Question

18. Provide a commitment in the FSAR stating that all required piping restraints, components and component supports have been installed in the piping systems prior to testing.

Response: Paragraph 14.2.4.1.2 indicates that certain test prerequisites must be satisfied prior to the initiation of any preoperational test. System lineup tests (SLTs) which require, as part b of paragraph 14.2.4.1.2 states, that pipe support inspections and adjustments be completed are examples of these preoperational test prerequisites. In addition a separate, distinct SLT governing verification of proper installation and adjustment of component supports has been generated. Execution of applicable portions of this SLT on each piping system provides formal documentation of required support operability.

The administrative frame work imposed upon the preoperational test program as described in FSAR Chapter 14 provides a commitment which requires that all required piping restraints, components and component supports have been installed prior to testing. In summary, sufficient discussion presently exists in the FSAR to address concerns in this area.

Summation - This item is closed.

Question

19. The applicant's preoperational test program covers the vibration and dynamic effects. However, the thermal expansion effects required in SRO 3.9.2.II-1.d, e and f are not adequately addressed. The thermal motion monitoring program should deal specifically with verification of snubber movement, adequate clearances and gaps to allow free movement of the pipe during heatup and cooldown and should include acceptance criteria and test procedures. Additional information on this program is required.

Response: The WNP-2 Thermal Expansion Program is conducted during the Startup Test Program which is described in FSAR section 14.2. The specific thermal expansion program is described in section 14.2.12.3.17. This section prescribes test purposes, prerequisites, a test description and acceptance criteria. This program will be applied to systems which experience an operating temperature greater than 250°F and are classified in one of the following categories:

- ASME Code Class 1, 2 or 3 piping system
- High energy piping system inside Seismic Category 1 structures
- High energy system whose failure could reduce the functioning of a Seismic Category 1 feature to an unacceptable safety level
- Seismic Category 1 portions of moderate energy piping system located outside containment
- Condensate/feedwater piping per Reg. Guide 1.68.1 c.2.f&g

A combination of visual inspection and remote monitoring of certain inaccessible locations on critical piping will provide data to make evaluations which address the defined test purposes. Specifically on selected systems, a pre-heatup visual inspection to establish baseline test conditions is performed which confirms that no potential obstruction thermal movement exists, pipe hangers are at their "cold positions", snubbers are at the mid-range of travel and adequate pipe whip restraint clearance exists. At an intermediate point and again at rate temperature, a visual examination of the selected piping systems is performed to confirm proper thermal movement relative to the baseline conditions. At corresponding reactor system temperatures, data is also recorded from the remote monitoring devices and compared against test acceptance criteria. Following several heatup and cooldown cycles, the thermal movement measurements are recorded a second time to determine that proper "shakedown" of the systems has occurred. Appropriate action based upon the test results is taken which includes a review of the system performance by the responsible piping design engineering organization and issuance of their findings.

During the visual inspections, special attention is directed to the following areas of piping/reactor system support components:

- Pipe whip restraint to pipe clearance at rated temperature
- Snubber expected movement and swing clearances at various temperatures including rated
- Control rod drive support structure to CRD housing gap at rated temperature
- Main steam piping penetration guide movement at rated temperature
- Reactor vessel seismic supports operability

  vessel to sacrificial shield stabilizers
  sacrificial shield to biological shield stabilizers

- Safety related process instrument piping movement such as:

  Reactor Vessel Level instrument piping
  Main Steam Flow instrument piping
  RCIC Steam Flow instrument piping

- Hot pipe containment penetration temperature profiles

The remote monitoring locations have not been finalized at present. Piping systems to be monitored have been tentatively identified that include the main steam, recirculation, feedwater, reactor core isolation cooling and, safety relief valve discharge line piping. The actual locations and selected piping systems will be established after an iterative selection process which consists of an assessment of the most advantageous measurement locations coupled with a review of possible monitoring locations. Both the responsible piping design organization and the Startup Test organization will thus cooperate to achieve an effective piping thermal movement monitoring program.

The finalized, detailed test procedure which delineates selected piping systems, applicable test acceptance criteria, visual inspection techniques, remote monitoring locations and required test conditions will be available on site for NRC inspection 60 days prior to commencement of the Startup Test Program on a schedule consistent with the preparation of other Startup Test procedures.

Summation - The Supply System will provide a reference in 3.9.2 to Chapter 14 of the FSAR. This item is closed.

See revised FSAR page 3.9.22 (attached).

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

3.9.1.4.13 Balance of Plant Equipment

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

3.9.2 DYNAMIC TESTING AND ANALYSIS

3.9.2.1 Preoperational Vibration and Dynamic Effects Testing on Piping

The test program is divided into three phases: preoperational vibration, startup vibration, and operation transients. Refer to Section 14.2.12.3.17 for a discussion of the piping thermal expansion test program.

3.9.2.1.1 Preoperational Vibration Testing

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

QUESTION NO. 20
(3.9.2.1)

The applicant has not given a clear description of the acceptance criteria for steady-state piping vibrat ons. The staff's position is that acceptance limi-s for vibration should be based on half the endurance limit as defined by the ASME Code at 10^6 cycles.

RESPONSE

For steady-state vibration, the piping peak stress due to vibration only (neglecting pressure) will not exceed 10,000 psi for Level 1 criterion and 5,000 psi for Level 2 criterion. These limits are below the piping material fatigue endurance limits as defined in Design Fatigue Curves in Appendix I of ASME Code for 10^6 cycles. The definitions of Level 1 and Level 2 criteria are clarified in the text revision attached.

Summation - This item is closed.

The FSAR will be changed to quantify Level 1 and Level 2 Criteria as indicated ~~above.~~

NOTE: References in Chapters 3 and 14 will be verified.

See revised FSAR pages 3.9-24, 3.9-24a, and 3.9-25 (attached).

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8/19/81

amplitude of displacements and number of cycles per transient of the main steam and recirculation piping are measured and the displacements compared with acceptance criteria. The deflections are correlated with stresses to verify the pipe stresses remain within Code limits. Remote vibration and deflection measurements are taken during the following transients:

a. Recirculation pump starts;

b. Recirculation pump at 100% of rated flow;

c. Turbine stop valve closure at 100% power;

d. Manual discharge of each S/R valve at 1,000 psig and at planned transient tests that result in S/R valve discharge.

3.9.2.1.5 Test Evaluation and Acceptance Criteria

The piping response to test conditions are considered acceptable if the organization responsible for the stress report reviews the test results and determines that the tests verify that the piping responded in a manner consistent with the predictions of the stress report and/or that the tests verify that piping stresses are within Code limits. To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. These criteria, designated Level 1 and 2, are described in the following paragraphs.

3.9.2.1.5.1 Level 1 Criteria

If in the course of the tests, measurements indicate that the piping is responding in a manner that would make test termination prudent, the test is terminated. Level 1 criteria establishes bounds on movement that, if exceeded, make a test hold or termination mandatory. The limits on movement are based on maximum allowable Code stress limits.

3.9.2.1.5.2 Level 2 Criteria

Conformance with Level 2 criteria demonstrates that the piping is responding in a manner consistent with the stress report

3.9.2.1.5.1 Level 1 Criterion

Level 1 establishes the maximum limits for the level of pipe motion which, if exceeded, makes a test hold or termination mandatory.

If the Level 1 limit is exceeded, the plant will be placed in a satisfactory hold condition, and the responsible piping design engineer will be advised. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 limits are satisfied.

3.9.2.1.5.2 Level 2 Criterion

Level 2 specifies the level of pipe motion which, if exceeded, requires that the responsible piping design engineer be advised. If the Level 2 limit is not satisfied, plant operating and startup testing plans would not necessarily be altered. Investigations of the measurements, criteria, and calculations used to generate the pipe motion limits would be initiated. An acceptable resolution must be reached by all appropriate and involved parties, including the responsible piping design engineer. Depending upon the nature of such resolution, the applicable tests may or may not have to be repeated.

~~predictions. Failure to meet Level 2 criteria doe~ not mean that the piping response is unsatisfactory; it means that the system is not responding in accordance with theoretical predictions and fur^her analyses ~ased on test results is necessary. Level 2 criteria is intended to screen out test results that are consistent with predictio~ and need no analytical review from those that must be evaluated.~~

3.9.2.1.6 Corrective Actions

During the course of the tests, the remote measurements are regularly checked to determine compliance with Level 1 criteria. If trends indicate that Level 1 crit:ria may be violated, the measurements are monitored at :ore frequent intervals. The test is held or terminated as soon as Level 1 criteria is violated. As soon as possible after the test hold or termination, the following corrective actions will be taken:

a. Installation Inspection. A walkdown of the piping and suspension is made to identify any obstruction or improperly operating suspension components. If vibration exceeds criteria, the source of the excitation must be identified to determine if it is related to equipment failure. Action is taken to correct any discrepancies before repeating the test.

b. Instrumentation Inspection. The instrumentation installation and calibration are checked and any discrepancies corrected. Additional instrumentation is added, if necessary.

c. Repeat Test. If actions (a) and (b) identify discrepancies that could account for failure to meet Level 1 criteria, the test is repeated.

d. Resolution of Findings. If the Level 1 criteria is violated on the repeat test or no relevant discrepancies are identified in (a) and (b), the organization responsible for the stress report shall review the test results and criteria to determine if the test can be safely continued.

Item # 21 Snubber Testing

The Supply System's response to the letter from R. Tedesco to R. Ferguson "Preservice Inspection and Testing of Snubbers" dated March 6, 1981 is contained in the letter from J. Shannon to R. Tedesco, G02-81-313, "Preservice Inspection and Testing of Snubbers," dated September 24, 1981. This letter states that the Supply System will comply with all of the requirements contained in NRC letter of March 6, 1981.

Summation - This item is closed.

021

Washington Public Power Supply System

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

September 24, 1981
G02-81-313

Docket No: 50-397

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

Attention: Mr. R. L. Tedesco
Assistant Director for Licensing
Division of Licensing

Subject: PRESERVICE INSPECTION AND TESTING OF SNUBBERS

Reference: Letter, R. L. Tedesco to R. L. Ferguson, "Preservice Inspection and Testing of Snubbers for WNP Units 1 through 5," dated March 6, 1981.

Dear Mr. Tedesco:

The Supply System has reviewed the Preservice Inspection and Pre-Operational Testing requirements transmitted with the reference letter. It is the intent of WNP-2 to fully comply with the requirements as stated. The Preservice Examination is presently detailed in the WNP-2 Preservice Inspection Program Plan Section 9.3.1; all of the reference letter requirements are complied with. The Pre-Operational Testing requirements will be detailed in Chapter 14 of the FASR at a later date.

Very truly yours,

J W Shannon

J. W. Shannon
Director, Safety and Security

JWS:JEP:cd

cc: R. Auluck, NRC
O. K. Earle, B&R
R. W. Hernan, NRC
N. S. Reynolds, D&L
D. D. Tillson

DSER 3.9.2.5(MEB-22)

internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and Standard Review Plan Section 3.9.2, and satisfies the applicable requirements of General Design Criteria 1 and 4.

3.9.2.5 Dynamic Analysis of Reactor Internals under Faulted Conditions

The applicant has presented inadequate data to verify the mathematical models for the dynamic analysis. Specifically an explanation of the dynamic model is requested and justification of the statement that "Only motion in the vertical direction will be considered here; hence, under structural member can only have an axial load."

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Our review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity and functionability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, or earlier industry standards.

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

a. The loading combinations and stress limits used in the design of (1) all ASME Class 1, 2, and 3 systems, components, equipment and their supports, (2) all reactor internals, and (3) control rod drive components need to be clarified in the FSAR. Section 3.9.3.1 and the majority of Tables 3.9.2 (a) through 3.9.2 (a c) in the FSAR do not clearly define the loading combinations and stress limits. We will require a concise summary (preferably in table form) of this information. This summary should include a listing of all the loads which were considered for each service condition or load case plus the acceptance criteria. Appendix 110-1 to NRC Question 110.27 contains loading combinations and acceptance criteria

QUESTION NO. 22
(3.9.2.5)

The applicant has presented inadequate data to verify the mathematical models for the dynamic analysis. Specifically, the explanation of the dynamic model is requested and justification of the statement that, "only motion in the vertical direction will be considered here; hence, each structural member can only have an axial load."

RESPONSE

Because of the shroud design in a BWR, the core flow during normal operation or a LOCA transient is always upward axially. Therefore, a vertical axial-flow model with 12 nodes is adequate to dynamically analyze the RPV internals. The text description of this model is clarified as the attached.

Summation - This item is closed.

PCY:rf/45E10
9/21/81

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces (see Figures 3.9-8a and 3.9-8b), a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods will be determined from a ~~comprehensive~~ dynamic model of the RPV and internals ~~with 37 degrees of freedom~~. Only motion in the vertical direction will be considered here; hence, each structural member (between two mass points) can only have an axial load. Besides the real masses of the RPV and core support structures, account will be made for the water inside the RPV.

Typical curves of the variation of pressures during a steam line break are shown in Figures 3.9-8a and 3.9-8b. The accident analysis method is described in 3.9.5.2.

The time varying pressures are applied to the dynamic model of the reactor internals described above. Except for the nature and locations of the forcing functions and the dynamic model, the dynamic analysis method is identical to that described for seismic analysis and is detailed in 3.7.2.1.

The loads and load combinations acting upon the jet pumps and LPCI coupling are listed in 3.9.3.1.

12-node vertical

internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and Standard Review Plan Section 3.9.2, and satisfies the applicable requirements of General Design Criteria 1 and 4.

3.9.2.5 Dynamic Analysis of Reactor Internals under Faulted Conditions

The applicant has presented inadequate data to verify the mathematical models for the dynamic analysis. Specifically an explanation of the dynamic model is requested and justification of the statement that "Only motion in the vertical direction will be considered here; hence, under structural member can only have an axial load."

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Our review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity and functionability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, or earlier industry standards.

3.9.3.1 Loading Combinations, Design Transients and Stress Limits

MEB-23

a. The loading combinations and stress limits used in the design of (1) all ASME Class 1, 2, and 3 systems, components, equipment and their supports, (2) all reactor internals, and (3) control rod drive components need to be clarified in the FSAR. Section 3.9.3.1 and the majority of Tables 3.9.2 (a) through 3.9.2 (a c) in the FSAR do not clearly define the loading combinations and stress limits. We will require a concise summary (preferably in table form) of this information. This summary should include a listing of all the loads which were considered for each service condition or load case plus the acceptance criteria. Appendix 110-1 to NRC Question 110.27 contains loading combinations and acceptance criteria

SER 3.9.3.1

applicable to all of the above systems, components, equipment and supports. Table 3.6-5 of the WNP-2 "Plant Design Assessment for SRV and LOCA Loads" presents information which is not completely acceptable. We will require a commitment to the Appendix 110-1 mentioned above. In addition, we will require a clarification of the applicability of Table 3.6.5, i.e. are all of these loading combinations and acceptance criteria applicable to all of the systems, components, equipment, etc. discussed in the first paragraph above?

b. Several references are made in Table 3.9.2 (a) through 3.9.2 (ac) to allowable stresses for bolting. Specifically, what loading combinations and allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

c. The applicant has not yet responded to Question 110.27, Appendix 110-2, "Interim Technical Position - Functional Capability of Passive Piping Components."

d. The methods of combining responses to all of the loads requested in a. above is required. Our position on this issue for Mark II plants is outlined in NUREG0484, Revision 1, "Methodolcgy for Combining Dynamic Responses." However, since the primary containment for the WNP-2 plant is a free-standing steel pressure vessel and the plant is in a higher seismic zone, the staff will require that the criteria in Section 4 of NUREG-0484, Revision 1, "Criteria for Combinations of Dynamic Responses other than those of SSE and LOCA" be satisfied if the square root of the sum of the squares method of combining these responses is used. (Reference Regulatory Position E (2) in the enclosure to a letter from J. R. Miller, NRC to Dr. G. G. Sherwood, G.E., "Review of General Electric Topical Report NEDE-24010-P," dated June 19, 1980.) The conclusions of NUREG-0484 Revision 1 are based on the studies performed by GE in NEDE-24010-P and BNL in NUREG/CR-1330. The applicant must demonstrate that an SRSS combination of dynamic responses achieves the 84% non-exceedance probability level because of the differences in containment and seismic level which were not included in the earlier studies.

e. The note in Table 3.9-2 (a) of the FSAR states that NSSS components designed to the upset plant condition (normal operating loads + upset transients + .5 SSE) will meet the upset design condition limits without a fatigue analysis. It is the staff's position that for all ASME Class 1 components a fatigue analysis shall be performed for all loading conditions. The basis for deviating from this position should be provided for WNP-2. If the WNP-2 position on this issue is implicit in the letter from W. Gang to R. Bosnak, "G.E. Position on Fatigues Analysis," dated January 15, 1981, provide the information requested in the letter from R. Bosnak to W. Gang dated February 19, 1981.

f. The safety relief valve discharge piping and downcomers are ASME Class 2 and 3 components, a fatigue analysis is not required in their design by the ASME Section III Boiler and Pressure Vessel Code. A through wall leakage crack in these lines resulting from fatigue caused by SRV actuations and small LOCA conditions would allow steam to bypass the pressure suppression pool. This could result in an unacceptable overpressurization of the containment. We, therefore, require that the applicant perform a fatigue evaluation on these lines in accordance with the ASME Class 1 fatigue rules.

g. Table 3.9-1 specifies one OBE with 10 maximum load cycles per event in the table of plant events. SRP 3.7.3 requires the use of 5 OBEs with 10 maximum load cycles per event. Justification of this reduced number of OBEs is requested. Note - This justification was also requested in the review of Section 3.7.3.

h. Table 3.9-2 (a) lists the allowable general membrane stress for the emergency loading conditions as 1.5 S_m. ASME Section III Figure 3224-1 specifies this limit as the greater of 1.2 S_m or S_y. What is the validity of the usage of 1.5 S_m. Also, the 1.5 S_m listed is 42300 psi. 1.5 x 26700 = 40050.

This table also specifies one of the loads for the emergency condition as maximum credible earthquake (Design Basis Earthquake) and one of the loads for faulted conditions as maximum credible earthquake. These terms have

not been previously defined and utilized. Are these loadings the SSE loadings?

i. In Table 3.9-2 (a), it is noted that the support skirt and the shroud support legs have been evaluated for buckling, but the buckling limits are not specified. The applicant should discuss the applicability of the criteria in FSAR Section 3.9.3.4, "Component Supports" to this table.

j. It is stated in Table 3.9-2 (a) that for the RPV Support (Bearing plate), the allowable stress for emergency conditions is 1.5 x AISC allowable stresses and for faulted conditions 1.67 x AISC allowable stresses. The applicant should provide the basis for these numbers.

For the RPV stabilizer, the allowable stresses are also based on the AISC specification. The allowable stress for the ROD is shown as 34,000 psi. What is the basis for this number? For the faulted loading condition, the allowable stress is shown as the material yield strength. Why is the difference from the previous faulted allowable stress of 1.67 x AISC allowable stress?

k. Table 3.9-2 (b) shows the general membrane plus bending allowable stress for emergency conditions as 1.5 S_A where $S_A = 1.5\ S_m$ and for faulted conditions as 2 S_A. What is the basis for these numbers? The ASME Section III code Figure NB3224-1 specifies 1.8 S_m or 1.5 S_y for emergency and Table F1322.2-1 specifies, 2.4 S_m or 0.7 S_u for components and 1.5 S_m or 1.2 S_y for component supports, for faulted conditions.

l. Table 3.9-2 (e) shows the allowable for the emergency condition as $P_e \leq 3.0\ S_m$. What is the significance and validity of this equation?

m. Table 3.9-2 (i) Item 9, Hanger Bracket Combined Stress. In the method of analysis, it is stated that the load = $(W_B + W_C + W_D).33$ and that the multiplier (.33) is added as a safety factor specified on the purchase part drawing. Without being able to evaluate the intent of this analysis in detail, it appears that this factor results in using only 0.33 of the

SER 3.9.3.1

total weight to determine the stresses. Additional details of this analysis are requested.

n. Table 3.9-2 (n) lists the calculated stresses and allowable stress for the ECCS Pumps. The actual stress exceeds the allowable for the RHR suction nozzle. While the excess is small, it is not noted what stresses, normal, upset, emergency or faulted, are being computed, and what loads were considered in determining these stresses. Additional information on the stresses in this area is requested.

MEB-36

o. In the discussion of the nozzle loads for the RCIC Pump on Page 3.9-50, it is not clear how the equation,

MEB-36

$$\frac{F_i}{F_o} + \frac{M_i}{M_o} \leq 1$$

is to be applied. Is F_i to be the maximum of F_x, F_y and F_z and M_i to be the maximum of M_x, M_y and M_z? Clarification is requested on this point.

p. Table 3.9.2(s). Justification is required for the usage of the AISC for the source of the allowable stresses and the source of the 1.6 S factor as the allowable stress. An explanation is also requested for the allowable stress of 0.7 ULT being equal to 35000 psi. If the material is 6061-T6 aluminum as noted in note a, the ultimate strength per ASTM B308 is 38000 psi so the allowable would be 0.78(38000) = 26600 psi.

MEB-37

q. Table 3.9-2(w). An explanation is requested for the 1.5 S_m and 2.25 S_m emergency stress limits and the 2 S_m and 3 S_m faulted stress limits.

MEB-38

r. Table 3.9-2(y) does not present adequate information for evaluation. What is meant by stress limits for VI and VII, and what are the stresses being evaluated?

MEB-39

s. Table 3.9-2(aa). The stresses evaluated are the Normal and Upset and the faulted loading condition. Why is there no emergency loading condition for this component.

MEB-40

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures

3.9.3.1 Loading Combinations Design Transients and Stress Limits

Question 23

The loading combinations and stress limits used in the design of (1) all ASME Class 1, 2 and 3 systems, components, equipment and their supports, (2) all reactor internals and (3) control rod drive components need to be clarified in the FSAR. Section 3.9.3.1 and the majority of Tables 3.9.2(a) through 3.9.2(ac) in the FSAR do not clearly define the loading combinations and stress limits. We will require a concise summary (preferably in table form) of this information. This summary should include a listing of all the loads which were considered for each service condition or load case plus the acceptance criteria. Appendix 110-1 to NRC Question 110.27 contains loading combinations and acceptance criteria applicable to all of the above system, components, equipment and supports. Table 3.6-5 of the WNP-2 "Plant Design Assessment for SRV and LOCA Loads" presents information which is not completely acceptable. We will require a commitment to the Appendix 110-1 mentioned above. In addition, we will require a clarification of the applicability of Table 3.6-5, i.e., are all of the loading combinations and acceptance criteria in Table 3.6-5 applicable to all of the systems, components, equipment, etc., discussed in the first paragraph above.

Response:

The Table number 3.6-5 in the question appears to be in error. Table 3.5-5 appears to be the table to which the question refers.

See revised Table 3.5-5 of the WNP-2 "Plant Design Assessment for SRV and LOCA Loads" for load combinations and acceptance criteria for balance of plant (attached).

See Table Q23-1 for the load combinations and acceptance criteria for NSSS piping and equipment.

Summation - The effects of hydrodynamic loads listed in the load combination table will be documented in the New Loads update. This item is closed.

TABLE 3.5-5 (DAR Rev. 2)

LOAD COMBINATIONS AND ACCEPTANCE CRITERIA

FOR ASME CODE CLASS 1,2, and 3 BOP PIPING AND EQUIPMENT **

Load Cases	Load Combinations (1)(2)	Design Assessment Acceptance Criteria
1	P+D.W.	Normal (A)
2	N+ OBE $+SRV_{ONE}$	Upset (B)
3	N+ OBE $+SRV_{TWO}$	Upset (B)
4	N+ OBE $+SRV_{ALL}$	Upset (B)
5	N+ OBE $+SRV_{ADS}$ +SBA	Emergency * (C)
6	N+ OBE $+SRV_{TWO}$ +SBA	Emergency * (C)
7	N+ SSE $+SRV_{ADS}$ +SBA/IBA	Faulted * (D)
8	N+ SSE $+SRV_{TWO}$ +SBA/IBA	Faulted * (D)
9	N+ SSE $+SRV_{ONE}$	Faulted * (D)
10	N+ SSE $+SRV_{TWO}$	Faulted * (D)
11	N+ SSE $+SRV_{ALL}$	Faulted * (D)
12	N+ SSE +DBA	Faulted * (D)

(1) As required by the appropriate subsection, i.e., NB, NC or ND of ASME Section III, Division 1, other loads, such as thermal transient, thermal gradients, and anchor point displacement portion of the OBE or SRV, may require consideration in addition to those primary stress-producing loads listed.

(2) SBA, IBA, and DBA include all event induced loads, as applicable, such as chugging, pool swell, drag loads, annulus pressurization, etc.

*All ASME Code Class 1, 2 and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's memorandum, "Evaluation of Topical Report - Piping Functional Capability Criteria", dated July 17, 1980.

**Equipment includes pumps, valves, supports, vessels. For bolting used in connection with the support of ASME Code Class 1,2, and 3 components, vendor load capacity data sheets are used or where design is by the architect engineer, stress levels are maintained less than specified minimum yield at temperature.

LOAD DEFINITION LEGEND (Table 3.5-5)

Normal (N)	-	Normal loads include internal pressure and dead weight
OBE	-	Operational Basis Earthquake loads
SSE	-	Safe Shutdown Earthquake loads
SRV_{TWO}	-	Safety/relief valve diacharge induced loads from two adjacent valves
SRV_{ALL}	-	The loads induced by actuation of all safety/relief valves
SRV_{ADS}	-	The loads induced by the actuation of safety/relief valves associated with the automatic depressurization system
SRV_{ONE}	-	The loads induced by the actuation of one safety/relief valve
SBA	-	Small Break Accident
IBA	-	Intermediate Break Accident
DBA	-	Design Basis Accident

TABLE Q23-1

LOAD COMBINATION AND ACCEPTANCE CRITERIA FOR ASME CODE CLASS 1, 2, AND 3 NSSS PIPING AND EQUIPMENT

23

Load Combination	Design Basis	Evaluation Basis	(Service Level)
N + $SRV_{(ALL)}$	Upset	Upset	(B)
N + OBE	Upset	Upset	(B)
N + OBE + $SRV_{(ALL)}$	Emergency	Upset	(B)
N + SSE + $SRV_{(ALL)}$	Faulted	Faulted*	(D)
N + SBA + SRV	Emergency	Emergency*	(C)
N + IBA + SRV	Faulted	Faulted*	(D)
N + SBA + $SRV_{(ADS)}$	Emergency	Emergency*	(C)
N + SBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + IBA + OBE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
N + SBA/IBA + SSE + $SRV_{(ADS)}$	Faulted	Faulted*	(D)
**N + LOCA + SSE	Faulted	Faulted*	(D)

LOAD DEFINITION LEGEND

Normal(N) - Normal and/or abnormal loads depending on acceptance criteria.

OBE - Operational basis earthquake loads.

SSE - Safe Shutdown earthquake loads.

SRV - Safety/relief valve discharge induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).

SRV_{ALL} - The loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV_{ADS} - The loads induced by the actuation of safety/relief valves associated with Automatic Depressurization System which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.

Note: This table will be inserted into the FSAR.

LOAD COMBINATION TABLE (CONT'D)

LOCA — The loss of coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).

$LOCA_1$ — Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.

$LOCA_2$ — Pool swell impact loads on piping and components located above the suppression pool water upper surface.

$LOCA_3$ — Oscillating pressure induced loads on submerged piping and components during condensation oscillations.

$LOCA_4$ — Building motion induced loads from chugging.

$LOCA_5$ — Building motion induced loads from main vent air clearing.

$LOCA_6$ — Vertical and horizontal loads on main vent piping.

$LOCA_7$ — Annulus pressurization loads.

SBA — The abnormal transients associated with a Small Break Accident.

IBA — The abnormal transients associated with an Intermediate Break Accident.

* All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's "~~Interim Technical Position-Function~~ Capability of ~~passive components~~" - by MEB.

** The most limiting case combination among $LOCA_1$ through $LOCA_7$.

QUESTION NO 24
(3.9.3.1)

Several references are made in Table 3.9.2(a) through 3.9.2(ac) to allowable stresses for bolting. Specifically, what loading combinations and allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flange connections. Where are these limits defined?

RESPONSE

1. Floor Mounted Equipment

(A) Equipment Anchorage Bolting

The floor anchored mechanical equipment (pumps, heat exchangers, and RCIC turbine) in GE's scope of supply are mounted on a concrete floor or a steel structure. The design of concrete anchor bolts for the equipment mounted on concrete floor, and the responsibility to prescribe and meet the necessary codes and stress limits are in the AE's scope of supply. The design of attachment bolts for the equipment mounted on steel structure, and the responsibility to prescribe and meet the necessary codes and stress limits are also in the AE's scope of supply. GE works with the interface limit of 10,000 psi in tension or shear for the only purpose of sizing bolt holes in the equipment base, based on the required nominal size and number of bolts for maximum loads.

(B) Component Support Bolting

(a) RWCU Pump

The support bolting of this non-safety essential pump is designed for the effects of pipe load and SSE load to the requirements of the ASME code, Section III, Appendix XVII. The stress limits of 0.41Sy for tension and 0.15Sy for shear are used.

(b) RCIC Turbine

The pump-to-base plate bolting is designed as follows:

(1) Normal Plus Upset

a) Primary membrane: 1.0S

b) Primary membrane plus bending: 1.5S, where S is the allowable stress limit per the ASME Code Section III, Appendix I, Table I-7.3.

(2) Emergency or Faulted

Stresses shall be less than 1.2 times the allowable limits for "Normal plus Upset" given above.

(C) Flanged Connection Bolting

There are no flange type connections in component supports.

2. Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports

The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D.

NF-3280 is applicable to bolting for Service Levels A and B.

For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum yield stresses at temperature.

Summation - This item is closed.

See revised FSAR pages 3.9-71, 3.9-158, and 3.9-159 (attached).

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(2) Snubbers were
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Bolting for Piping Supports and Pipe Mounted Equipment (Valves and Pump) Supports

The supports are hanger and snubber type (including clamps) linear standard components as defined by the ASME Code Section III, Subsection NF. The bolts used in these supports meet criteria of NF-3280 for Service Levels A and B and NF-3230 for Service Levels C and D. NF-3280 is applicable to bolting for Service Levels A and B.

For Service Levels C and D, XVII-2460 with factors indicated under XVII-2110 is applicable to the design requirements of bolting. The calculated stresses under these categories do not exceed the specified minimum yield stresses at temperature.

Add these words as new item e.

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of the above abnormal environ-
sient test, the snubber shall be
dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.

d. Rigid Supports

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

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

3.9.3.4.2 ECCS Pumps

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

TABLE 3.9-2 (p)

RWCU PUMP

Following is a summary of the design calculations on the RWCU Pump:

Pump Part (ASME Code Calculation)	Calculated Stress (psi)	Allowable Stress (psi)
Casing wall	10,476	12,814
Suction wall	5,112	12,814
Discharge wall	3,337	12,814
Cover bolting	23,032	30,750
Seal gland bolting	26,532	30,750
Pedestal bolt (shear)	18,015	44,000
Motor Part		
Motor foot bolts (shear)	3,787	44,000

The support bolting of this non-safety essential pump is designed for the effects of pipe load and SSE load to the requirements of the ASME code, Section III, Appendix XVII. The stress limits of 0.41Sy for tension and 0.15Sy for shear are used.

Add this.

TABLE 3.9-2 (q)

RCIC TURBINE

The following is a summary of the design calculations on the RCIC turbine components:

	Stress (psi) Calculated	Stress (psi) Allowable
Pressure Boundary Castings		
Stop valve	9,800 psi	14,000 psi
Governor valve	13,200 psi	17,500 psi
Turbine inlet (high press.)	15,300 psi	21,000 psi
Turbine wheel case (low press.)	18,000 psi	21,000 psi
Pressure Boundary Bolting		
Stop valve	20,100 psi	25,000 psi
Governor valve	16,510 psi	25,000 psi
Turbine flange	13,410 psi	25,000 psi
Non-Pressure Boundary Components		
Turbine shaft	11,450 psi	50,000 psi
Thrust bearing	1,250 lbf	1,550 lbf
Journal bearing	575 lbf	1,390 lbf
Stop valve yoke	7,475 psi	36,000 psi
Pedestal dowel pins	46,880 psi	61,100 psi
Pedestal bolts	11,900 psi	32,000 psi

The pump-to-base plate bolting is designed as follows:

(1) Normal Plus Upset

a) Primary membrane: 1.0S

b) Primary membrane plus bending: 1.5S, where S is the allowable stress limit per the ASME Code Section III, Appendix I, Table I-7.3.

(2) Emergency or Faulted

Stresses shall be less than 1.2 times the allowable limits for "Normal plus Upset" given above.

Add this.

3.9.3.1 Loading Combinations Design Transients and Stress Limits

Question 25

The applicant has not yet responded to Question 110.27, Appendix 110-2, "Interim Technical Position - Functional Capability of Passive Piping Components."

Response:

BOP

Piping system functional capability is being evaluated using the criteria given in NRC memorandum, "Evaluation of Topical Report - Piping Functional Capability Criteria," dated July 17, 1980.

NSSS

Referring to the response to Q. 110.27, the WNP-2 project does comply with Appendices A and B to Section 110. Accordingly, the statement of compliance is shown as a footnote in the attached load combination table.

Summation - This item is closed.

See FSAR revised pages 3-LIV, 3.9-95a and 3.9-95b (attached).

LIST OF TABLES (Continued)

TABLE 3.9-2 - General)

LOAD COMBINATION AND ACCEPTANCE CRITERIA
FOR ASME CODE CLASS 1, 2, AND 3
NSSS PIPING AND EQUIPMENT

Load Combination	Design Basis	Evaluation Basis	(Service Level)
$N + SRV_{(ALL)}$	Upset	Upset	(B)
$N + OBE$	Upset	Upset	(B)
$N + OBE + SRV_{(ALL)}$	Emergency	Upset	(B)
$N + SSE + SRV_{(ALL)}$	Faulted	Faulted*	(D)
$N + SBA + SRV$	Emergency	Emergency*	(C)
$N + IBA + SRV$	Faulted	Faulted*	(D)
$N + SBA + SRV_{(ADS)}$	Emergency	Emergency*	(C)
$N + SBA + OBE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
$N + IBA + OBE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
$N + SBA/IBA + SSE + SRV_{(ADS)}$	Faulted	Faulted*	(D)
** $N + LOCA + SSE$	Faulted	Faulted*	(D)

LOAD DEFINITION LEGEND

Normal(N) - Normal and/or abnormal loads depending on acceptance criteria.

OBE - Operational basis earthquake loads.

SSE - Safe Shutdown earthquake loads.

SRV - Safety/relief valve discharge induced loads from two adjacent valves (one valve actuated when adjacent valve is cycling).

SRV_{ALL} - The loads induced by actuation of all safety/relief valves which activate within milliseconds of each other (e.g., turbine trip operational transient).

SRV_{ADS} - The loads induced by the actuation of safety/relief valves associated with Automatic Depressurization System which actuate within milliseconds of each other during the postulated small or intermediate size pipe rupture.

Table 3.9-2 - General (CONT'D)

LOCA	—	The loss of coolant accident associated with the postulated pipe rupture of large pipes (e.g., main steam, feedwater, recirculation piping).
$LOCA_1$	—	Pool swell drag/fallout loads on piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	—	Pool swell impact loads on piping and components located above the suppression pool water upper surface.
$LOCA_3$	—	Oscillating pressure induced loads on submerged piping and components during condensation oscillations.
$LOCA_4$	—	Building motion induced loads from chugging.
$LOCA_5$	—	Building motion induced loads from main vent air clearing.
$LOCA_6$	—	Vertical and horizontal loads on main vent piping.
$LOCA_7$	—	Annulus pressurization loads.
SBA	—	The abnormal transients associated with a Small Break Accident.
IBA	—	The abnormal transients associated with an Intermediate Break Accident.

* All ASME Code Class 1, 2, and 3 piping systems which are required to function for safe shutdown under the postulated events shall meet the requirements of NRC's ~~"Interim Technical Position-Function Capability of passive components"~~ - by MEB.

** The most limiting case combination among $LOCA_1$ through $LOCA_7$.

QUESTION No. 26
(3.9.3.1)

The methods of combining responses to all of the loads requested in (a) above is required. Our position in this issue for Mark II plants is outlined in NUREG-0484, Revision 1, "Methodology for Combining Dynamic Responses". However, since the primary containment for the WNP-2 plant is a free-standing steel pressure vessel and the plant is in a higher seismic zone, the staff will require that the criteria in Section 4 of NUREG-0484, Rev. 1, "Criteria for Combination of Dynamic Responses other than those of SSE and LOCA" be satisfied if the square root of the sum of the squares method of combining these responses is used. (Reference Regulatory Position E (2) in the enclosure to a letter from J. R. Miller, NRC to Dr. G. G. Sherwood, GE, "Review of General Electric Topical Report NEDE-24010-P", dated June 19, 1980). The conclusions of NUREG-0484, Rev. 1 are based on the studies performed by GE in NEDE-24010-P and BNL in NUREG/CR-1330. The applicant must demonstrate that an SRSS combination of dynamic responses achieves the 84% nonexceedance probability level because of the difference in containment and seismic level which were not included in the earlier studies.

RESPONSE

When a seismic response from a high seismic input, like that from Hanford, is combined with another dynamic response (e.g. SRV discharge loads), depending on the relative magnitudes of the two responses being combined, the shape of the cumulative Distribution Function (CDF) of the combined response will change. If the maximum magnitude of one of the responses is very large compared to the other response being combined, the CDF curve will almost be vertical and it is immaterial if these two responses are combined using the SRSS or the Absolute Sum (ABS) rule. However, if the maximum magnitudes of the two responses are about equal, use of SRSS vs. ABS rule to combine the responses will with cause significant difference in the combined response. In addition, in this case, the CDF curve will be more like S-shaped with the non-exceedance probability (NEP) of SRSS being close to 84%. In the generic Mark II study, examples from both such cases were considered with more examples from the case with responses of comparable magnitudes. This study showed that all these Mark II cases meet the requirements of the NUREG-0484. Hence, the GE topical report NEDE-24010-P, "Technical Bases for the Use of SRSS Method for Combining Dynamic Loads for Mark II Plants" is also applicable to WNP-2 with high seismic input.

The impact of the free-standing steel primary containment is discussed in the areas as follows:

(1) Vessel and Internals are not attached to and not affected by the steel containment.

(2) Piping Systems and Floor Mounted Equipment

The dynamic input to these components at their containment support locations may be affected by the steel containment response to the dynamic loads under consideration and hence, may be different from that obtained from concrete containment. However, the frequencies contributing to the responses of major structures and components in both types of plants will not be significantly different but will fall into the same general range.

The structural frequencies will only determine the magnitude of amplification or attenuation of the response. For multi-frequency random-type dynamic loads, the components of input loads whose frequencies coincide with the structural natural frequencies will be amplified and these components will dominate the response. Although the predominant response of a particular structural component may vary somewhat in frequency between the concrete and steel containment configuration, the variances are expected to be small for the range of frequencies of interest for major structures because of the similarities in systems, types of structural configurations, construction materials and massiveness of buildings. Therefore, key characteristics of the responses (duration of strong response motion and number of peaks) are primarily determined by the input component loads to the structure, and because of the similarity of the dynamic nature of the input loads due to earthquake, SRV and LOCA for both types of containment, their structural responses will have similar dynamic characteristics. Hence, the response of the mechanical components and piping systems supported from the two types of containments will also be similar. Hence, the use of SRSS combinations for combining the dynamic responses for the WNP-2 application will be demonstrated to meet the 84% non-exceedance probability level.

Summation: - This item is closed.

See FSAR revised pages 3-xLi and 3.9-72 (attached).

TABLE OF CONTENTS (Continued)

3.9.3.4.3 RCIC Turbine

The RCIC turbine assembly has been tested, as defined in 3.9.2.2. These tests proved the adequacy of the support structure for the turbine assembly under actual operating conditions. Furthermore, the calculation summary provided in 3.9.3.1 defined the stress levels in the critical support areas, namely, the stop valve yoke and the pedestal dowel pins and bolts. The substantial stress level margins prove the adequacy of the equipment.

3.9.3.4.4 Reactor Water Cleanup System Pump

The pump and pedestal bolts have been analyzed as discussed in 3.9.3.1.15(c). Loads from seismic dead weight, connecting pipes, and temperature were considered.

3.9.3.5 Containment Mounted Equipment (INSERT ATTACHED)

3.9.4 CONTROL ROD DRIVE SYSTEM (CRDS)

This plant is equipped with a hydraulic control rod drive system. The discussion in 3.9.4 includes the control rod drive mechanism (CRDM), the hydraulic control unit (HCU), the condensate supply system and the scram discharge volume and extends to the coupling interface with the control rods.

3.9.4.1 Descriptive Information Regarding CRDS

Descriptive information on the control rod drives as well as the entire control and drive system is contained in 4.6.

3.9.4.2 Applicable CRDS Design Specifications

The control rod drive system (CRDS) is designed to meet the functional design criteria as outlined in 4.6 and consists of the following:

a. Locking piston control rod drive;

b. Hydraulic control unit;

c. Hydraulic power supply (pumps),

d. Interconnecting piping,

NEW INSERT FOR 3.9.3.5

When a seismic response from a high seismic input, like that from ~~Hanford,~~ WNP-2 is combined with another dynamic response (e.g. SRV discharge loads), depending on the relative magnitudes of the two responses being combined, the shape of the cumulative Distribution Function (CDF) of the combined response will change. If the maximum magnitude of one of the responses is very large compared to the other response being combined, the CDF curve will almost be vertical and it is immaterial if these two responses are combined using the SRSS or the Absolute Sum (ABS) rule. However, if the maximum magnitudes of the two responses are about equal, use of SRSS vs. ABS rule to combine the responses will cause significant difference in the combined response. In addition, in this case, the CDF curve will be more like S-shaped with the non-exceedance probability (NEP) of SRSS being close to 84%. In the generic Mark II study, examples from both such cases were considered with more examples from the case with responses of comparable magnitudes. This study showed that all these Mark II cases meet the requirements of the NUREG-0484. Hence, the GE topical report NEDE-24010-P, "Technical Bases for the Use of SRSS Method for Combining Dynamic Loads for Mark II Plants" is also applicable to WNP-2 with high seismic input.

The impact of the free-standing steel primary containment is discussed in the areas as follows:

(1) Vessel and Internals are not attached to and not affected by the steel containment.

(2) Piping Systems and Floor Mounted Equipment

The dynamic input to these components at their containment support locations may be affected by the steel containment response to the dynamic loads under consideration and hence, may be different from that obtained from concrete containment. However, the frequencies contributing to the responses of major structures and components in both types of plants will not be significantly different but will fall into the same general range.

The structural frequencies will only determine the magnitude of amplification or attenuation of the response. For multi-frequency random-type dynamic loads, the components of input loads whose frequencies coincide with the structural natural frequencies will be amplified and these components will dominate the response. Although the predominant response of a particular structural component may vary somewhat in frequency between the concrete and steel containment configuration, the variances are expected to be small for the range of frequencies of interest for major structures because of the similarities in systems, types of structural configurations, construction materials and massiveness of buildings. Therefore, key characteristics of the responses (duration of strong response motion and number of peaks) are primarily determined by the input component loads to the structure, and because of the similarity of the dynamic nature of the input loads due to earthquake, SRV and LOCA for both types of containment, their structural responses will have similar dynamic characteristics. Hence, the response of the mechanical components and piping systems supported from the two types of containments will also be similar. Hence, the use of SRSS combinations for combining the dynamic responses for the WNP-2 application will be demonstrated to meet the 84% non-exceedance probability level.

[illegible]

QUESTION NO. 27
(3.9.3.1)

The note in Table 3.9-2(a) of the FSAR states that NSSS components designed to the upset plant condition (normal operating loads + upset transients & .5 SSE) will meet the upset design condition limits without a fatigue analysis. It is the staff's position that for all ASME Class 1 components a fatigue analysis shall be performed for all loading conditions. The basis for deviating from this position should be provided for WNP-2. If the WNP-2 position on this issue is implicit in the letter from W. Gang to R. Bosnak, "GE Position on Fatigue Analysis," dated January 15, 1981, provide the information requested in the letter from R. Bosnak to W. Gang, dated February 19, 1981.

RESPONSE

The information requested was documented in the letter from R. B. Johnson to R. Bosnak, "GE Position on Fatigue Analysis," on June 29, 1981. A copy is attached.

Summation - This item is closed.

PCY:rf/45E12
8/18/81

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

MFN 122-81

ENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-3297

June 29, 1981

Mr. R. Bosnak, Chief
Mechanical Engineering Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Bosnak:

SUBJECT: GE POSITION ON FATIGUE ANALYSIS

Reference: Letter, R. Bosnak to W. G. Gang, same subject, dated February 19, 1981

This letter responds to the referenced letter requesting that GE provide assurance that the methodologies employed to evaluate fatigue effects properly considers the combination of the OBE and SRV loads. GE's approach to fatigue evaluations is clarified as follows:

In the fatigue analysis of NSSS equipment, piping, reactor pressure vessel and RPV internal components, the actual calculated loads due to OBE and SRV are combined to show compliance with upset limits for fatigue. This calculation is performed by comparing the "plant unique OBE and SRV loads" with the "original OBE load used for the design basis." If the "plant unique OBE and SRV load" exceeds the "original OBE load used for the design basis," a stress evaluation is done to show the stresses to be within acceptance limits. The larger of the two loads has been evaluated for 10 or more fatigue cycles consistent with upset limits. For reactor vessel nozzle loads, the original OBE load is also the maximum permissible value shown in the interface control document (ICD) issued by General Electric. Consequently, OBE loads have been combined with other upset loads (including SRV) for the fatigue evaluation.

The procedure described above is applied in general to all BWR 4/5/6 requisition projects. The actual calculated loads (OBE + SRV) are more commonly used for BWR 4/5 projects, but in either case, a comparison is made to insure that the ICD loads are not exceeded.

The number of SRV cycles used for these calculations varies widely for BWR 4/5 projects. However, the number of SRV cycles for BWR/6 projects is always less than 13000 because of the low-low set feature which is part of the standard BWR/6 design.

This approach has been discussed with you and members of your staff and we understand it is acceptable.

Very truly yours,

R. B. Johnson

R. B. Johnson, Acting Manager
BWR Projects Licensing
Nuclear Safety and Licensing Operation

RBJ:sem/1125-26 625

cc: L. S. Gifford

bcc: R. Villa
G. G. Sherwood
P. C. Yin
BWRPL Staff

3.9.3 ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations Design Transients and Stress Limits

Question 28

The safety relief valve discharge piping and downcomers are ASME Class 2 and 3 components, a fatigue analysis is not required in their design by the ASME Section III Boiler and Pressure Vessel Code. However, a through wall leakage crack in these lines resulting from fatigue caused by SRV actuations and small LOCA conditions would allow steam to bypass the pressure suppression pool. This could result in an unacceptable overpressurization of the containment. We, therefore, require that the applicant perform a fatigue evaluation on these lines in accordance with the ASME Class 1 fatigue rules.

Response:

A fatigue evaluation using ASME Class 1 fatigue rules is currently being performed for the downcomers and the wetwell portion of the SRV piping potentially subject to bypass leakage.

Summation - The results of this evaluation will be reported in the WNP-2 Design Assessment Report for SRV and LOCA loads. This item is closed.

QUESTION NO. 29
(3.9.3.1)

Provide justification for utilizing one OBE with 10 maximum load cycles specified in Table 3.9-1.

RESPONSE

The justification is provided in the response to Question No. 9. Revision to Table 3.9-1 is attached to the response to Question No. 10.

Summation - This item is closed.

See FSAR revised pages 3.9-92 and 3.9-93 (attached).

PCY:ggt:rf/45L24
9/23/81

TABLE 3.9-1

PLANT EVENTS

	No. of Cycles
Normal, Upset, and Testing Conditions	
a. Bolt Up*/Unbolt	123
b. Design Hydrostatic Test	130
c. Startup (100°F/hr Heatup Rate)**	120
d. Daily Reduction to 75% Power*	10,000
e. Weekly Reduction to 50% Power*	2,000
f. Control Rod Pattern Change*	400
g. Loss of Feedwater Heaters (80 Cycles Total):	80
h. Operating Base Earthquake Event at Rated Operating Conditions	~~5~~**** 10/50
i. Scram:	
1) Turbine Generator Trip, Feedwater on, Isolation Valves Stay Open	40
2) Other Scrams	140
3) Loss of Feedwater Pumps, Isolation Valves Closed	10
4) Single Safety or Relief Valve Blowdown	8
j. Reduction to 0% Power, Hot Standby, Shutdown (100°F/hr Cooldown Rate)**	111
k. HPCS Operation (10), SLC Operation (10)	20

TABLE 3.9-1 (Continued)

		No. of Cycles
Emergency Conditions		
1.	Scram:	
	1) Reactor Overpressure with Delayed Scram, Feedwater Stays on, Isolation Valves Stay Open	1***
	2) Automatic Blowdown	1***
m.	Improper Start of Cold Recirculation Loop	1***
n.	Sudden Start of Pump in Cold Recirculation Loop	1***
o.	Improper Startup with Reactor Drain Shut Off Followed by Turbine Roll and Increase to Rated Power	1***
Faulted Condition		
p.	Pipe Rupture and Blowdown	1***
q.	Safe Shutdown Earthquake at Rated Operating Conditions	1***
ASME Hydrostatic Test		
r.	1.25 x Design Pressure Hydrostatic Test (per NB 6222 and NB 3114)	10

*Applies to reactor pressure vessel only.

**Bulk average vessel coolant temperature change in any 1-hour period.

***The annual encounter probability of the one cycle events is $<10^{-2}$ for emergency and $<10^{-4}$ for faulted events.

****Includes ~~10 maximum load cycles per event.~~ 50 peak OBE cycles for NSSS piping and all BOP piping and components, and 10 peak OBE cycles for all NSSS equipment and components.

QUESTION NO. 30
(3.9.3.1)

Provide the basis for utilizing the allowable general membrane stress for the emergency loading conditions as 1.5 Sm in Table 3.9-2(a). ASME Section III Figure 3.2.2.4-1 specifies this limit as the greater of 1.2 Sm or Sy. This table also specifies one of the loads as maximum credible earthquake which has not been clearly defined.

RESPONSE

The listed stress criterion is in typographical error. "1.5 Sm" should be replaced with Sy. See the table revision attached. The maximum credible earthquake is SSE.

TABLE 3.9-2 (a) (Continued)

Vessel Support Skirt

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B and PVC Sect. III Primary Stress Limit for SA 533 GRB CL1 For normal and upset Condition:	Normal and upset condition loads	General Membrane	26,700	19,911
S_m = 26,700 psi	1. Dead Weight 2. Design earthquake (Operating basis earthquake)			
For emergency condition:	Emergency condition loads	General Membrane	42,300	39,245
$\cancel{1.5S_m}$ Sy = 42,300 psi	1. Dead Weight 2. Maximum credible earthquake ~~(Design basis earthquake)~~ (SSE)			
For faulted condition:	Faulted condition loads	General Membrane	42,300	39,245
$\cancel{1.5S_m}$ Sy = 42,300 psi	1. Dead Weight 2. Maximum credible earthquake (SSE) 3. Jet reaction forces			

NOTES: The vessel support skirt has been evaluated for buckling.

Faulted category loads were evaluated with emergency allowable loads.

30

QUESTION NO. 31
3.9.3.1

In Table 3.9-2(a), it is noted that the supported skirt and shroud support legs have been evaluated for buckling, but the buckling criteria are not specified. The applicant should discuss the applicability of the criteria in FSAR Section 3.9.3.4, "Component Supports" to this table.

RESPONSE:

(a) The response to Question 42 addresses the subject of support skirt buckling.

(b) The criterion for the shroud support, which is a core support structure is defined by Equation b in Table 3.9-9 of the FSAR. The maximum faulted condition design load is 854.5 kips per shroud support leg compared to a critical buckling load of 1289 kips. A copy of Table 3.9-9 is attached for reference.

Summation - This item is closed.

See revised FSAR pages 3.9-99 and 3.9-100 (attached).

TABLE 3.9-9

BUCKLING STABILITY LIMIT

(for reactor internal structures only)

Any One Of (No More Than One Required)	General Limit
a. $\left[\dfrac{\text{Permissible load, LP}}{\text{Code normal event permissible load, PN}}\right]$	$\le \dfrac{2.25}{SF_{min}}$
b. $\left[\dfrac{\text{Permissible load, LP}}{\text{Stability analysis load, SL}}\right]$	$\le \dfrac{0.9}{SF_{min}}$

where

LP = permissible load under stated conditions of normal, upset, emergency or fault.

PN = applicable code normal event permissible load.

SL = stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

TABLE 3.9-2 (a) (Continued)

Vessel Support Skirt

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B and PVC Sect. III Primary Stress Limit for SA 533 GRB CL1				
For normal and upset Condition:	Normal and upset condition loads	General Membrane	26,700	19,911
S_m = 26,700 psi	1. Dead Weight 2. Design earthquake (Operating basis earthquake)			
For emergency condition:	Emergency condition loads	General Membrane	42,300	39,245
$1.5S_m$ = 42,300 psi	1. Dead Weight 2. Maximum credible earthquake (Design basis earthquake)			
For faulted condition:	Faulted condition loads	General Membrane	42,300	39,245
$1.5S_m$ = 42,300 psi	1. Dead Weight 2. Maximum credible earthquake 3. Jet reaction forces			

NOTES: The vessel support skirt has been evaluated for buckling. (See 3.9.3.4)

Faulted category loads were evaluated with emergency allowable loads.

TABLE 3.9-2 (a) (Continued)

Shroud Support Legs

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
ASME B and PVC Sect. III Primary Local Membrane Plus Primary Bending Limit for SA-533 Grade B Class 1 For Design Mechanical Load condition:	Design Mechanical Load	Local Membrane Plus Bending	40,050	22,890
$1.5\ S_m = 40,050$ psi	1. Dead Weight 2. Design earthwuake (Operating basis earthquake)			
For emergency condition: $1.5 \times S_y = 63,450$ psi	Emergency condition loads 1. Dead Weight 2. Maximum credible earthquake (Design basis earthquake)	Local Membrane Plus Bending	63,450	32,240
For faulted condition:	Faulted condition loads	Local Membrane Plus Bending	63,450	32,240
$1.5 \times S_y = 63,450$ psi	1. Dead Weight 2. Maximum credible earthquake. 3. Jet reaction forces 4. Pressure drop across core support plate and shroud head			

NOTE: The shroud support legs have been evaluated for buckling. Maximum faulted condition design load is 854.5 kips per leg compared to a critical buckling load of 1289 kips. (See Table 3.9-9.) Faulted category loads were evaluated with emergency allowable stresses.

QUESTION NO. 32
(3.9.3.1)

Provide the basis for utilizing the allowable stress for emergency condition of 1.5 x AISC allowable stresses and for faulted conditions of 1.67 x AISC allowable stresses for the RPV support (bearing plate). For the RPV stabilizer, the allowable stresses are also based on the AISC specification. The allowable stress for the rod is shown as 84,000 psi. What is the basis for this number? For the faulted loading condition, the allowable stress is shown as the material yield strength. Why is the difference from the the previous faulted allowable stress of 1.67 x AISC allowable stress?

RESPONSE

1. Bearing Plate

(a) Faulted Condition

GE Report NEDE-10949-3 and GESSAR establish the basis for the 1.5 x AISC allowable for supports and structures. Since AISC = 2/3 of yield strength for bending, it follows that, for A-36 material,

1.5 x AISC = 1.5 x (2/3 of yield strength) = yield strength = 36,000 psi

(b) Normal and Upset Condition

Two thirds of yield is 24,000 psi, but 22,000 psi is used for conservatism.

(c) Emergency Condition

This condition is not critical to an inactive equipment, therefore, a 1.5 factor is applied to the normal and upset limit to arrive at the emergency limit.

The above clarifications are added to Table 3.9-2(a) as footnotes.

2. RPV Stabilizer

The rod yield strength is 140,000 psi which is used as the faulted limit. Based on the AISC criterion for tension, 0.6 x 140,000 = 84,000 psi is used for normal and upset. Accordingly, the table entry is clarified by the added footnotes.

Summation - This item is closed.

See revised FSAR pages 3.9-103 and 3.9-104 (attached).

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TABLE 3.9-2 (a) (Continued)

RPV Support (Bearing Plate)

Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit				
AISC specification for the design, fabrication and erection of structural steel for buildings.	Normal and upset condition 1. Dead loads 2. Operating basis earthquake 3. Loads due to scram	Bearing Plate	22,000[(1)]	f_b = 8,000
For normal & upset conditions AISC allowable stresses, but without the usual increase for earthquake loads.				
For emergency conditions 1.5 x AISC allowable stresses. (2)	Emergency condition 1. Dead loads 2. Design basis earthquake 3. Loads due to scram	Bearing Plate	33,000	f_b = 16,000
For faulted conditions ~~1.67 x AISC allowable stresses for structural steel members.~~ yield strength of A-36 steel.	Faulted condition 1. Dead loads 2. Design basis earthquake 3. Jet reaction load	Bearing Plate	36,000	f_b = 16,800

(1) $2/3\ S_y$ = 24,000 psi; 22,000 psi used for added conservatism.

(2) The factor of 1.5 is applied to the normal and upset limit since the emergency condition is not critical for inactive equipment.

TABLE 3.9-2 (a) (Continued)

RPV Stabilizer

Criteria	Loading	Location	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit				
AISC specification for the construction, fabrication, and erection of structural steel for buildings	Upset condition	Rod	84,000 (1)	f_t = 54,000
	1. Spring preload	Bracket	22,000	f_b = 22,000
	2. Operating basis earthquake	Bracket	14,000	f_v = 4,600
For normal & upset contions AISC allowable stresses, but without the usual increase for earthquake loads				
For emergency conditions 1.5 x AISC allowable stresses	Emergency condition	Bracket	33,000	f_b = 24,400
	1. Spring preload	Bracket	21,000	f_v = 10,600
	2. Design basis earthquake	Rod	126,000 (2)	f_t = 108,000
For faulted conditions Material yield strength	Faulted condition			
	1. Spring preload	Bracket	36,000	f_b = 26,000
	2. Design basis earthquake	Bracket	21,500	f_v = 11,330
	3. Jet reaction load	Rod	140,000	f_t =132,000

(1) 0.6 Sy based on the AISC criterion for tension.

(2) 1.5 times the normal and upset limit.

QUESTION NO. 33
(3.9.3.1)

Table 3.9-2(b) shows the general membrane plus bending allowable stress for emergency conditions as $1.5S_A$ where S_A = 1.5 Sm and for faulted conditions as 2 S_A. What is the basis for these numbers? The ASME Section III code Figure NB3224-1 specifies 1.8 Sm or 1.5 Sy for emergency and Table F1322.2-1 specifies, 2.4 Sm or 0.7 Su for components and 1.5 Sm or 1.2 Sy for component supports, for faulted conditions.

RESPONSE

1. For emergency conditions, the 2.25 Sm limit is same as the limit per ASME Subsection NG.

2. For faulted conditions, the 3.0 Sm limit is more conservative than the 3.6 Sm value in Appendix F, Table F1322.2-1 as shown by the comparison below:

From Appendix F, $P_m \leq$ 0.7 Su or 2.4 Sm

P (membrane + bending)		
≤ 1.5 x 0.7 Su	or	1.5 x 2.4 Sm
= 1.05 Su	or	3.6 Sm
= 1.05 x 63,500	or	3.6 x 16,925
= 66,675 psi	or	60,930 psi

Hence, in either case, the limit of 50,775 psi is Table 3.9-2(b) is more conservative than Appendix F.

An error in the stress type is corrected as attached.

Summation - This item is closed.

PCY:ggt:rf/45L14
9/23/81

TABLE 3.9-2 (b)

REACTOR VESSEL INTERNALS AND ASSOCIATED EQUIPMENT

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
TOP GUIDE-HIGHEST STRESSED BEAM				
Primary Stress Limit The allowable primary membrane stress plus bending stress is based on ASME Boiler and Pressure Vessel Code, Sect. III for type 304 stainless steel plate.				
For normal and upset condition Stress Intensity $S_A = 1.5\ S_m = 1.5 \times 16{,}925$ psi = 25,388 psi	Normal and upset condition loads 1. Operating basis earthquake 2. Weight of structure	General Membrane Plus Bending	25,388	21,676
For emergency condition: $S_{limit} = 1.5\ S_A = 1.5 \times 25{,}338 = 38{,}081$ psi	Emergency condition loads 1. Design basis earthquake 2. Weight of structure	General Membrane Plus Bending	38,081	32,514
For faulted condition: $S_{limit} = 2\ S_A = 2 \times 25{,}208 = 50{,}775$ psi	Faulted condition loads (same as emergency condition)	General Membrane Plus Bending	50,775	32,514

TO MATCH OTHER 3.9.2 TABLES, AMD-2

QUESTION NO. 34
(3.9.3.1)

Table 3.9-2(e) shows the allowable for the emergency condition as Pe $\leq$ 3.0 Sm. What is the significance and validity of this equation?

RESPONSE

The criterion "Pe $\leq$ 3.0 Sm" should be deleted. "Eq. 9 $\leq$ 2.25 Sm" is the criterion for both emergency load cases. Accordingly, Table 3.9-2(e) is revised as attached.

In the new loads update for BOP, Tables 3.9-16 and 3.9-17 will be upgraded to cover piping, components and supports.

1 TABLE 3.9-2 (e)

CLASS ~~(I)~~ RECIRCULATION LOOP PIPING

LOADING COMBINATIONS AND STRESS LIMITS

Loading Combinations	Allowables
DESIGN	
$P_D + W + OBE_I$	Eq. 9 $\le 1.5\ S_m$ (NB-3652)
NORMAL/UPSET	
P_o, W, OBE_I, OBE_D, TE	U < 1.0 Eq. 12 $\le 3.0\ S_m$ Eq. 13 $\le 3.0\ S_m$ (NB-3653)
EMERGENCY	
$P_o + W + OBE_I$	$\cancel{P_e \le 3.0\ S_m}$ Eq. 9 $\le 2.25\ S_m$ (NB-3655)
$P_e + W$	Eq. 9 $\le 2.25\ S_m$ (NB-3655)
FAULTED	
$P_o + W + SSE_I$	Eq. 9 $\le 3.0\ S_m$ (F-1360 Appendix F)
TEST	
$P_t + W$	$P_m \le 0.9\ S_y$
P_t	Eq. 9 $\le 1.35\ S_y$ (NB-3226)

QUESTION NO. 35
(3.9.3.1)

Table 3.9-2 (i) Item 9, Hanger Bracket Combined Stress. In the method of analysis, it is stated that the load = $(W_B + W_C + W_D)$ .33 and that the multiplier (.33) is added as a safety factor specified on the purchase part drawing. Without being able to evaluate the intent of this analysis in detail it appears that this factor results in using only 0.33 of the total weight to determine the stresses. Additional details of this analysis are requested.

RESPONSE

The recirculation pump is suspended from four hanger rods. The load on each rod should be $(W_B + W_C + W_D) \times 0.25$. In the actual design, $(W_B + W_C + W_D) \times 0.33$ is assumed. This provides a 32% safety margin.

This is clarified by the footnote in the attached table revision.

Summation - This item is closed.

TABLE 3.9-2 (1) (Continued)

Criteria	Method of Analysis	Analytical Results	Allow. Stress or Actual Thickness
7. Seal Gland Retainer	$S_s = \frac{W}{w_{dt}}$	S_s = 5486 psi	S_s = 9480 psi
A. Loads:			
Normal and upset condition	w = load imposed d = diameter at shear resistance		
Design pressure & temperature	t = thickness at shear resistance		
B. Allowable working stress per ASME Code Sect. VIII.			
8. Shock Suppressor Lug Combined Stress	Loads shall be applied in the normal direction simultaneously to determine tensile, shear and bending stresses in the brackets. Tensile, shear, and bending stresses shall be combined to determine max. combined stresses.	Combined Stress (Shear plus Tensile) Lug #1 S_C = 21,430 psi Lug #2 S_C = 20,915 psi Lug #3 S_C = 15,540 psi	S_m = 19435 psi S_y = 21,600 psi
A. Loads:			
DBE horizontal seismic force = 1.5 g			
B. Combined Stress Limit:			
Yield stress per ASME Sect. III			
9. Hanger Bracket Combined Stress	Bracket vertical loads shall be determined by summing the equipment and fluid weights and vertical seismic forces.	S_C = 8,327 psi	S_m = 12,600 psi
A. Loads:	(1)		
Flooded weight of equipment	Load = $(W_B + W_C + W_D).33$		
DBE vertical seismic force = 0.14 g	~~Note: the multiplier (.33) is added as a safety factor specified on the Purchase Part Drawing.~~		
B. Combined Stress Limit:	W_B = weight of motor		
Yield stress per ASME Sect. VIII	W_C = weight of motor mount W_D = weight of pump case		

(1) There are four hanger brackets. The load on each should be total load divided by 4. The use of total load divided by 3 gives a 33% safety factor.

QUESTION NO. 36
(3.9.3.1)

Table 3.9-2(n) lists the calculated stresses and allowable stress for the ECCS Pumps. The actual stress exceeds the allowable for the RHR suction nozzle. While the excess is small, it is not noted what stresses, normal, upset, emergency or faulted, are being computed, and what loads were considered in determining these stresses. Additional information on the stresses in this area is requested.

In the discussion of the nozzle loads for the RCIC Pump on page 3.9-50, it is not clear how the equation,

$$\frac{Fi}{Fo} + \frac{Mi}{Mo} \leq 1$$

is to be applied. Is Fi to be the maximum of Fx, Fy and Fz and Mi to be the maximum of Mx, My and Mz? Clarification is requested on this point.

RESPONSE

1. RHR Suction Nozzle Stress

Table 3.9-2 (n) has been updated and replaced by three comprehensive sub-tables. The requested additional information on the stresses is provided in details shown as attached.

2. RCIC Pump Nozzle Loads

The clarification is provided in the attached text revision.

Summation - This item is closed.

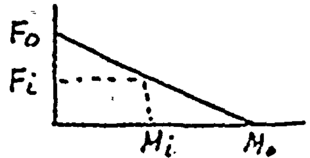
See revised FSAR pages 3.9-50, 3.9-155, 3.9-155a, and 3.9-155b (attached).

Note vertical bars added to equation

g. Nozzle Loading:

Pump nozzles are subject to loading from the connecting pipe. The nozzle pipe reactions to the allowable forces and moments on the equipment is expressed as:

$$\left|\frac{Fi}{Fo}\right| + \left|\frac{Mi}{Mo}\right| \leq 1 \qquad \cancel{Fi = Fx = Fy = Fz} \quad \cancel{Mi = Mx = My = Mz}$$



Fo = The allowable value of Fi when all moments are zero; and

Mo = The allowable value of Mi when all forces are zero. Therefore, the equipment shall be designed to be capable of:

a. Withstanding the three external orthogonal forces, all equal to Fo with no moments.

b. Withstanding the three external orthogonal moments, all equal to Mo with no forces.

Table 3.9-2(r) contains a summary of the design calculation for the RCIC pump components.

3.9.3.1.11 ECCS Pumps

Design condition for RHR, LPCS, and HPCS pumps are as follows:

	RHR	LPCS	HPCS
Design pressure			
Suction	220 psig	100 psig	100 psig
Discharge	500 psig	550 psig	1715 psig
Design Temperature	40-360°F	40-212°F	40-212°F

F_i = largest of the three external orthogonal forces (F_x, F_y, and F_z) imposed by the pipe.

M_i = largest of the three external orthogonal moments (M_x, M_y, and M_z) permitted from the pipe when they are combined simultaneously for a specific condition.

TABLE 3.9-2 (n)

ECCS PUMPS

The following is a summary of the design calculations on pump components:

	Calculated Stress (psi) RHR	LPCS	HPCS	Allowable (psi)
Pressure Boundary Parts				
Suction shell	18756	11025	11345	21000
Discharge nozzle	8040	8040	12060	17500
Suction nozzle	27383	14246	14248	27000
Torispherical head of shell	10365	4711	5139	17500
Stuffing box	2028	2230	7847	15000
Nozzle head lower plate	9635	2516	11582	15000
Mech. seal press. bolting	7600	7600	13660	25000
Mounting flange	11293	9838	5846	17500
Nozzle bolting	20978	15676	16545	25000
Non-Pressure Boundary Components				
Motor mounting bolting	21075	18259	12693	25000
Motor mounting flange	860	153	8946	17500

DELETE. REPLACE WITH 3 NEW PAGES (ATTACHED).

TABLE 3.9-2(m)

ECCS PUMPS

RESIDUAL HEAT REMOVAL PUMP

Suction Barrel Shell

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) ~~ACTUAL THICKNESS (INS.)~~	ALLOWABLE STRESS (PSI) ~~MIN. THICKNESS (INS.)~~
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	6,399	20,400
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	13,680	18,000
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	19,029	34,650
Discharge Elbow Discharge Elbow or Suction Pipe (max.)	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	10,643	21,600
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME, Section III Subsection NF	2,996	15,200
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME, Section III Subsection NF	6,081	17,500

~~TABLE 3.9-2n (continued)~~ TABLE 3.9-2(n) (Continued)

ECCS PUMPS

LOW PRESSURE CORE SPRAY PUMP

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) ~~OR ACTUAL THICKNESS (IN.)~~	ALLOWABLE STRESS (PSI) ~~OR MIN. THICKNESS (IN)~~
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III.	9,037	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III.	11,355	15,000
Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	12,170	34,650
Discharge Elbow or Suction Pipe (max)	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	8,758	17,500
Motor Stand	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME, Section III., Subsection NF	2,623	15,200
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME, Section III, Subsection NF	4,824	17,500

~~TABLE 3.9-3n - (Continued)~~ TABLE 3.9-2(n) (Continued)

ECCS PUMPS

HIGH PRESSURE CORE SPRAY PUMP

LOCATION	LOADING CONDITION	CRITERIA	CALCULATED STRESS (PSI) ~~OR ACTUAL THICKNESS (IN.)~~	ALLOWABLE STRESS (PSI) ~~OR MIN. THICKNESS (IN.)~~
Suction Barrel Shell	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	5,115	21,000
Stuffing Box Pipe	Design Pressure Static Loads Dynamic Load	ASME Boiler & Pressure Vessel Code, Section III	12,851	18,000
Suction Nozzle Shell Inter Section	FAULTED CONDITION Design Pressure Static Loads Dynamic Load	ASME Boiler & Pressure Vessel Code, Section III	13,733	39,650
Discharge Elbow or Suction Pipe (Max.)	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	ASME Boiler & Pressure Vessel Code, Section III	12,499	21,000
Motor Stand	FAULTED CONDITION Design Pressure Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME Section III, Subsection NF	1,348	15,200
Motor Bolting	FAULTED CONDITION Static Loads Dynamic Loads	Bolting Loads & Stresses per ASME Section III, Subsection NF	3,821	21,000

QUESTION NO. 37
(3.9.3.1)

Table 3.9-2(s). Justification is required for the usage of the AISC for the source of the allowable stresses and the source of the 1.6 S factor as the allowable stress. An explanation is also requested for the allowable stress of 0.7 ULT being equal to 35000 psi. If the material is 6061-T6 aluminum as noted in note a, the ultimate strength per ASTM 8308 is 38000 psi so the allowable would be 0.78(38000) = 26600 psi.

Response

1. Justification of AISC

   The spent fuel storage racks are designed by a different contractor than are the new fuel storage racks. Both contractors used the AISC since the fuel racks are structural devices, not pressure-retaining systems.

   The 1.6 S factor is permitted for factored load conditions by Standard Review Plan 3.8.4, Section II.4.

   The spent fuel racks are 300 series stainless steel, the new fuel racks are aluminum.

2. Allowable Stresses

   The limit of 0.7 Fu is not used. A new table is provided using a factor of 1.33 to raise the normal allowable for the upset allowable in accordance with AISC, Part 1, Section 1.5.6. The upset allowable is then used for the emergency and faulted conditions as shown in the new table attached.

Summation - This is a revised response. Item is closed.

TABLE 3.9-2 (s)

FUEL STORAGE RACKS

CRITERIA	LOADING	LOCATION	ALLOWABLE STRESS (0.7 ULT)	CALCULATED STRESS
1. NEW FUEL STORAGE RACKS	FAULTED CONDITION "A"			
Stress due to normal upset or emergency loading shall not cause a failure so as to result in a critical array	1. Dead Loads 2. Full Fuel Load in rack 3. S.S.E. 4. Thermal (not applicable)	1. Beam (Axial) 2. Beam (Trans.) 3. Combined	1. 35,000 $\#/in^2$ 2. 35,000 $\#/in^2$ 3. 35,000 $\#/in^2$	1. 15,090 $\#/in^2$ 2. 6,673 $\#/in^2$ 3. 16,500 $\#/in^2$

NOTES:

Source of Allowable Stress (0.7 ULT)

a. ASTM B308 Alloy 6061-T6
b. ASME Code - Boilers and Pressure Vessels, Sect. III, NA
c. Product Safety Standards for BWR-6-Mark III, Sect. VI, A. (3).
d. ASME - Pressure Vessels and Piping: Design and Analysis, Volume One, Page 69.
e. ASTM code for Boilers and Pressure Vessels was selected on the premise that data used from this source would necessarily be on the conservative side as applied to the fuel storage rack calculations.

Delete. Replace with the attached.

AMENDMENT NO. 4
June 1979
Page 1 of 2

1. NEW FUEL STORAGE RACKS

ACCEPTANCE CRITERIA	LOADING	PRIMARY STRESS TYPE	ALLOWABLE STRESS (psi)	CALCULATED STRESS (psi)
The allowable stress is based on Part 1 of AISC Manual for type ASTM B221, 6061-T6 Alum. Alloy F_u = 38,000 psi F_y = 35,000 psi				
For normal condition: S_{limit} = .66 F_y	For normal condition: Normal operating loads	Axial Load + Bending	23,100	15,230
For emergency condition:(1),(2) S_{limit} = .88 F_y	For emergency condition: Normal operating loads Operating Basis Earthquake Safety Relief Valve LOCA	Axial Load + Bending	30,800	30,800
For faulted condition: S_{limit} = .88 F_y	For faulted condition: Normal operating loads Safe Shutdown Earthquake Safety Relief Valve LOCA	Axial Load + Bending	30,800	30,800

(1) A one-third margin is added to the normal limit to obtain the upset limit per AISC, 7th Edition, Part 1, Section 1.5.6

(2) The upset allowable is used to evaluate emergency and faulted conditions for conservatism.

TABLE 3.9-2 (s) (Continued)

SOURCE OF LOADS AND STRESSES

S.S.E. loads derived by dynamic analysis. Total stress refers to combined earthquake and thermal load at highest expected pool temperature. Earthquake stresses obtained by square root of the sum of the squares method for a response due to tri-axial excitation. Stress given is the highest in the total structural array.

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AMENDMENT NO. 4
June 1979
Page 2 of 2

QUESTION NO. 38
(3.9.3.1)

Table 3.9-2(w). An explanation is requested for the 1.5 Sm and 2.25 Sm emergency stress limits and the 2 Sm and 3 Sm faulted stress limits.

RESPONSE

The current Table 3.9-2(w) is superseded by a new table which provides the stress limits on the basis of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. The FSAR text description of jet pumps is also revised accordingly.

Summation - This item is closed.

PCY:ggt:rf/45L18
9/23/81

3.9.1.4.2 Standard Reactor Interior Components

3.9.1.4.2.1 CR Guide Tube

The maximum calculated stress on the CR guide tube occurs in the base during an SSE and is 19,654 psi. The faulted limit is the lesser of 2.4 Sm or 0.7 Su at the design temperature per ASME Code, Section III, Table I-1.2 and F 1322-1. The faulted condition loads are shown on Table 3.9-2(aa). The faulted condition stresses are within elastic limits and are also shown on this table.

3.9.1.4.2.2 Incore Housing

The faulted condition maximum calculated stress on the Incore Housing occurs at the outer surface of the vessel penetration during a SSE and is 15,290 psi. The allowable stress for the elastic analysis used is Sm = 20,000 psi and the ultimate strength of the material is 57,500 psi. Table 3.9-2(ab) shows the faulted loads applied. The stresses are within elastic limits.

3.9.1.4.2.3 Jet Pump

The elastic analysis for the jet pump faulted conditions shows that the maximum stress occurs at the riser brace ~~is due to impulse loading of the diffuser during a pipe rupture and blowdown~~ and is 54,450 ~~39,500~~ psi. The maximum allowable for this condition per ASME code Section III, Subsection NG, is 3.6Sm or 60,840 psi. ~~is 3 Sm or 60,000 psi. The maximum stress under faulted loading conditions at any point in the jet pump other than that discussed above is approximately 2,500 psi.~~ Table 3.9-2(w) shows the ~~faulted~~ loads ~~applied.~~ summary.

3.9.1.4.2.4 LPCI Coupling

The maximum stress during a SSE on the LPCI coupling occurs at the "bellows" which is a purchased component designed to GE requirements for 120 normal operating condition cycles and 10 SSE cycles. The stresses on the bellows are within elastic limits.

3.9.1.4.2.5 Orificed Fuel Support

Due to its complex configuration, a series of vertical and horizontal load tests were performed on the orificed fuel support (OFS) in order to verify the design. Results from these tests indicate that the component and seismic loading

TABLE 3.9-2H

JET PUMPS

CRITERIA	LOADING COMBINATIONS	STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
PRIMARY MEMBRANE PLUS BENDING STRESS BASED ON ASME BSPV CODE SECTION III, SUB SECTION NG.				
FOR SERVICE LEVELS A & B (NORMAL AND UPSET) CONDITION: FOR TYPE 304 S.S. @ 550°F S_m = 16,800 psi S_{limit} = 1.5 S_m psi	Normal Loads(1) OBE SRV	PRIMARY MEMBRANE PLUS BENDING	25,200	6,618
FOR SERVICE LEVEL C (EMERGENCY) CONDITION: FOR TYPE 304 S.S. @ 550°F S_m = 16,800 psi S_{limit} = 2.25 S_m psi	Normal Loads(1) LOCA SRV	PRIMARY MEMBRANE PLUS BENDING	37,800	6946
FOR SERVICE LEVEL D (FAULTED) CONDITION: FOR TYPE 304 S.S. @ 550°F S_m = 16,800 psi S_{limit} = 3.6 S_m psi	Normal Loads(2) LOCA SSE	PRIMARY MEMBRANE PLUS BENDING	60,840	54,450(3) ~~32,232~~

TO MATCH OTHER 3.9.2 TABLES

RWH

(1) Design internal pressure, hydraulic and pressure reaction loads.

(2) Design external pressure, hydraulic and pressure reaction loads.

(3) Riser brace only. Stresses on other components are much lower.

3.9-168

QUESTION NO. 39
(3.9.3.1)

Table 3.9-2(y) does not present adequate information for evaluation. What is meant by stress limits for VI and VII, and what are the stresses being evaluated?

RESPONSE

Table 3.9-2(y) is revised as attached.

Summation - This item is closed.

PCY:ggt/45L16
9/22/81

WNP-2

TABLE 3.9-2 (y)

LPCI COUPLING

CRITERIA	LOADING COMBINATIONS	STRESS TYPE	ALLOWABLE STRESS (PSI)	CALCULATED STRESS (PSI)
PRIMARY MEMBRANE PLUS BENDING STRESS BASED ON ASME B&PV CODE SECTION III FOR TYPE 316L STAINLESS STEEL FOR SERVICE LEVELS A & B (NORMAL & UPSET) CONDITION: $S_{LIMIT} = 3S_m = 41,850$ psi	Normal + OBE + SRV_{ALL}	PRIMARY (and secondary) MEMBRANE + BENDING *	41,850	13,455
FOR SERVICE LEVEL C (EMERGENCY) CONDITION: $S_{LIMIT} = 2.25S_m = 31,400$ psi	Normal + LOCA (chugging) + SRN_{ADS} SRV_{ADS}	PRIMARY MEMBRANE + BENDING	31,400	22,938
FOR SERVICE LEVEL D (FAULTED) CONDITION: $S_{LIMIT} = 3.6S_m = 50,220$ psi	Normal + SSE + LOCA (Annulus Pressurization)	PRIMARY MEMBRANE + BENDING	50,220	33,660

TO MATCH OTHER 3.9-2 TABLES

Cwix

* Excluding thermal bending per NB-322.8.3. 3228.3

QUESTION NO. 40
(3.9.3.1)

Table 3.9.2(aa). The stresses evaluated are Normal and Upset and the faulted loading condition. Why is there no emergency loading condition for this component.

RESPONSE

For control rod guide tube, there is no emergency load condition.

Summation - This item is closed.

See revised FSAR page 3.9-173 (attached).

TABLE 3.9-2 (aa)

CONTROL ROD GUIDE TUBE

Criteria	Loading *(1)(2)	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
CONTROL ROD GUIDE TUBE				
Primary Stress Limit - The allowable primary membrane stress plus bending stress is based on the ASME Boiler and Pressure Vessel Code, Section III for type 304 stainless steel tubing	Normal and upset condition applied loads 1. External pressure 2. Vertical seismic + weight 3. Horizontal seismic 4. Lateral flow impingement 5. Vibration	Applying vertical seismic plus dead weight the maximum stress under normal & upset condition occurs at the guide tube base.		
For normal and upset conditions: $S_m = 16,000$ psi				
For faulted condition: $S_{limit} = 2.4S_m = 2.4 \times 16,000 = 38,400$ psi	Faulted condition applied loads 1. External pressure 2. Vertical seismic + weight 3. Horizontal seismic 4. Lateral flow impingement 5. Vibration	Applying vert. seismic plus dead weight the maximum stress under faulted loading conditions occurs at the guide tube base	24,000 38,400	14,745 19,654

(1) Dynamic loads are added directly

(2) There is no emergency load condition for these tubes.

DSER 3.9.3.2

Equipment Qualification Branch
Input for Safety Evaluation Report
WNP-2

3.9.3.2 Pump and Valve Operability Assurance

The staff has reviewed the applicant's pump and valve operability assurance program as discussed in Section 3.9.3.2 of the FSAR and compared this information with Section 3.9.3 of the Standard Review Plan. Based on our review, the applicant has provided information to define how active pumps and valves are generally qualified with respect to operability.

However, and in particular, for those components where qualification and/or operability assurance is by analysis alone, some question remains as to the confidence level assured by this methodology. The necessity for additional component testing is being considered and can not be established without an inspection at the plant site. Therefore, for the staff to determine the adequacy of the implementation of the applicant's pump and valve operability assurance program, an on-site audit of the equipment and supporting documentation is required.

The on-site audit will include a plant inspection to observe the as-built configuration and installation of the equipment. Also during the audit the staff will review qualifying documentation, eg., test reports, and analysis, which are described in the applicants program. Thus our overall review includes an FSAR review and an on-site audit of the equipment. Both phases of the staff review must be determined acceptable to arrive at a favorable conclusion on the applicant's overall pump and valve operability assurance program.

The applicant had been requested to provide information on the completion status of the equipment documentation, and on-site installation of the equipment. Before the audit is conducted, 85 to 90 percent completion should be attained for both the equipment documentation and the on-site installation of the equipment.

Once the applicant has indicated that his work is substantially complete, the staff will conduct an on-site audit shortly thereafter.

Because of the limited number of equipment that can be audited within a reasonable time, the audit results must provide a high degree of confidence that the implementation of the applicant's program is acceptable.

DSER 3.9.3.3 (MEB-41)

3.9.3.3 Design and Installation of Pressure Relief Devices

We have reviewed the design and installation criteria applicable to the mounting of pressure relief devices used for the overpressure protection of ASME Class 1, 2, and 3 safety and relief valves. We have specifically reviewed the applicant's compliance with SRP 3.9.3.

The response to Question 110.031 in the FSAR, Amendment 9 does not comply with the guidelines in Regulatory Guide 1.67, "Installation of Overpressure Devices" concerning dynamic load factor. Paragraph 3.9.3.3.2, "Open Relief Systems," implies that there may be pressure relief devices of the WNP-2 plants which relieve to open discharge systems. More information on what dynamic load factor was used and how it was determined is required.

In addition, the applicant is requested to provide a commitment that all of the information in Sections 3.9.3.3.2 and 3.9.3.3.3 of the FSAR are applicable to both NSSS and BOP supplied components.

Based upon our review of FSAR Section 3.9.3.3 and contingent upon the satisfactory resolution of the open items, our findings will be as follows:

The criteria used in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67 and Standard Review Plan Section 3.9.3.

3.9.3.3 Design and Installation of Pressure Relief Devices

Question 41

a. The response to Question 110.031 in the FSAR, Amendment 9, does not comply with the guidelines in Regulatory Guide 1.67, "Installation of Overpressure Devices" concerning dynamic load factor, Paragraph 3.9.3.3.2 of the FSAR, "Open Relief Systems", implies that there may be pressure relief devices of the WNP-2 plant which relieve to open discharge systems. More information on what dynamic load factor was used and how it was determined is required. In addition, the applicant is requested to provide a commitment that all of the information in Sections 3.9.3.3.2 and 3.9.3.3.3 of the FSAR are applicable to both NSSS and BOP supplied components.

b. Indicate how relief valve transients are treated. Clarify whether it is the intention of the FSAR to indicate that all relief valve transients are treated using detailed dynamic analysis techniques.

RESPONSE

a. See revised 3.9.3.3.2 of the FSAR. WNP-2 design does not include any open relief system, therefore, 3.9.3.3.2 has been deleted from the FSAR. Section 3.9.3.3.3 is applicable to both NSSS and BOP supplied components.

b. Relief valves which produce transient loadings are evaluated using detailed dynamic analysis techniques.

   1) Detailed dynamic analysis techniques are applied for the evaluation of the 18 mainsteam safety relief lines (See FSAR Section 3.9.3.3.1).

   2) Transient analyses for the relief valves listed below are performed using detailed dynamic analysis methods as described in FSAR Section 3.9.3.3.3.

      RHR-RV-95A
      RHR-RV-95B
      RHR-RV-55A
      RHR-RV-55B
      RHR-RV-36

   See revised 3.9.3.3 of the FSAR.

To clarify the FSAR, the attached revisions have been prepared.

Summation - This item is closed.

Qualification testing of sensitive electrical/pneumatic equipment to meet performance requirements defined in Tables 3.11-1, 3.11-2 and 3.11-3 is completed.

Seismic tests have been conducted on the safety relief valves and the natural frequencies have been determined to be $\geq$ 33Hz. The tests also determined that the equipment remains functional during application of the specified "G" loads.

In addition to testing described above and in 3.9.2.2.2, the sensitive electrical/pneumatic equipment of the safety/relief valve has been qualified to performance requirements during and after emergency environment conditions defined in Tables 3.11-1, 3.11-2 and 3.11-3.

The MSIV and S/RV (Safety/Relief Valve) analytical qualification results are shown in Tables 3.9-2(h) and 3.9-2(g) respectively.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

The design criteria for all safety and relief piping are in accordance with the rules in Subarticles NB-3677 and NC-3677 of ASME Section III, and the rules of Code Case 1569, applicable to the classification of the piping component under investigation. For relief systems the design criteria and the analyses used to calculate maximum stresses and stress intensities are in accordance with Subarticles NB-3600 and NC-3600 of ASME Section III. The maximum stresses are calculated based upon the full discharge loads, including the effects of the system dynamic response, and the system design internal pressure. Stresses are determined for all significant points in the piping system including the safety valve inlet pipe nozzle and the nozzle to shell juncture.

INSERT

3.9.3.3.1 Main Steam Safety/Relief Valves

Safety/relief valve lift results in a transient that produces momentary unbalanced forces acting on the discharge piping system for the period from opening of the safety/relief valve until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the discharge piping following the relatively rapid opening of the safety/relief valve cause the safety/relief valve discharge piping to vibrate. This in turn produces forces that act on the main steam piping.

Insert to 3.9.3.3.

Detailed evaluations are performed only for valves which produce transient effects; small relief valves (for example, those relieving temperature induced water expansion), where pressure relief is accomplished without transient effects, are not evaluated.

The analysis of the relief valve discharge transient consists of a stepwise time history solution of the fluid flow equation, to generate a time-history of the fluid properties at numerous locations along the pipe. Simultaneously, reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum change, and fluid friction terms. Figure 3.9-3 shows a set of fluid property and pipe section load transients typical of those produced by relief valve discharge.

The method of analysis applied to determine piping system response to relief valve operation is time history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the safety/relief valve, the main steam line, and the discharge piping are combined with loads due to other effects as specified in 3.9.3.1. The Code stress limits, corresponding to load combinations classification as normal, upset, emergency and faulted, are applied to the steam and discharge pipe.

3.9.3.3.2 Open Relief Systems

~~The total steady state discharge thrust load for an open dis-~~ charge system is expressed as the sum of the pressure and momentum forces as follows:

$$\frac{F}{A} = 144\,(P) + \frac{V^2\rho}{g}$$

where
F = Total Reaction Force lbf.
A = Exit Flow Area, ft^2
P = Exit Pressure, lbf/in^2 gage
V = Exit Fluid Velocity, ft/sec
ρ = Exit Fluid Density, lbm/ft^3
g = Gravity Acceleration, 32.2 $\frac{lbm\text{-}ft}{lbf\text{-}sec^2}$

To ensure consideration of the effects of the suddenly applied loads on the valve nozzle and pipe junction, a dynamic load factor is computed. The calculation of dynamic load factor is based on modeling the valve and nozzle as a single degree of freedom dynamic system. The lumped mass of this system corresponds to the weight of the valve and nozzle ~~and is assumed to be at the valve center of gravity. The~~

There are no open discharge pressure relief values mounted on ASME Class 1, 2, or 3 systems.

~~rotational degree of freedom of this system is considered to~~ be in the direction that causes maximum bending stress in the nozzle at the junction of the nozzle and run-pipe. Rotational flexibility of the system is computed by a series combination of nozzle flexibility and local run-pipe flexibility (at the junction of the nozzle and run-pipe).

The rise time of the discharge force at the outlet of the safety valve elbow is assumed to be the minimum valve opening time, and the discharge force is assumed to rise linearly with time. The ratio of maximum dynamic rotations predicted by this single degree of freedom system to the static rotation caused by the steady state discharge force represents the dynamic load factor.

To ensure the consideration of the effects of the suddenly applied loads on the pipe system, a dynamic time history analysis is performed on the piping system. The forcing function applied at the point of discharge is a linear force change from zero to the value of (F) that is determined in the above equation over a time period (t) that corresponds to the valve opening time which is provided by the valve manufacturer. After time (t) has been reached the force remains at the value of (F) until the conclusion of the time history integration. The lumped mass model that represents the piping system includes the safety-relief valves.

Where more than one valve is mounted on a common header, two cases are computed. In the first, full discharge of all valves is assumed to occur simultaneously. In the second the forcing functions are applied to a combination of valves that yields the worst load case. This worst load case is first ~~verified by trial through a series of static load cases.~~

3.9.3.3.3 Closed Relief System

For relief valve discharging into closed system, an analytical model of one-dimensional transient flow characteristics following the blow-off of the upstream safety/relief valve into the discharging piping system is established. The time-dependent pressure, temperature, density, velocity and hence the momentum of the downstream pipe flow are then computed from this conservative hydrodynamic/thermodynamic flow model. The phenomena such as flow restrictions, frictional resis-

DSER 3.9.3.4.a (MEB-42)

3.9.3.4 Component Supports

We have reviewed information submitted by the applicant relative to the design of ASME Class 1, 2, and 3 component supports. Our review included an assessment of the structural integrity of the supports and the effect of support deformation on the operability of active pumps and valves.

Our review has resulted in the following open issues:

a. The applicant's response to NRC Question 110.29 is not completely acceptable. The revised paragraph 3.9.3.4 states, "In design of the reactor vessel support skirt as a plate and shell-type component support, the allowable compressive load was limited to 90 percent of the load which produces a stress equivalent to yield stress in the material, divided by the safety factor for the plant condition being evaluated. The safety factor for the faulted condition was 1.125. The effects of fabrication and operational eccentricity were included in stress calculations." This implies that the reactor vessel support skirt was designed to an allowable compressive load of .8 material yield stress. It is not clear how the applicant's design would meet the staff's acceptable allowable load of two-thirds of critical buckling load. In addition, the applicant has assumed the critical buckling stress as the material yield stress at temperature. This definition could result in a non-conservative value for critical buckling stress. Critical buckling stress depends upon the configuration (including manufacturing effects) and the material properties (elastic modulus, E and minimum yield strength S_y) of the load bearing number. Because both of these material properties change with temperature, the critical buckling stress should be calculated using the values of E and S_y at the temperature.

The applicant will be required to provide the basis for using the critical buckling stress as defined in the FSAR and to clarify how the design of the reactor vessel support skirt meets the staff's acceptable allowable load of two-thirds of the critical buckling load.

QUESTION NO. 42
(3.9.3.4)

The applicant's response to NRC Question 110.29 is not completely acceptable. Paragraph 3.9.3.4 implies that the reactor vessel support skirt was designed to an allowable compressive load of .8 material yield stress. It is not clear how the applicant's design would meet the staff's acceptable allowable load of two-thirds of critical buckling load. In addition, the applicant has assumed the critical buckling stress as the material yield stress at temperature. Provide basis for this assumption.

RESPONSE

Per GE design specification, the permissible compressive load on the reactor vessel support skirt cylinder (plate and shell type component support) was limited to 90 percent of the load which produces yield stress, divided by the safety factor for the condition being evaluated. The effects of fabrication and operational eccentricity was included. The safety factor for faulted conditions was 1.125.

An analysis of reactor pressure vessel support skirt buckling for faulted conditions shows that the support skirt has the capability to meet ASME Code Section III, Paragraph F-1370(c) faulted condition limits of 0.67 times the critical buckling strength of the support at temperature. The faulted condition analyzed included the compressive loads due to the design basis maximum earthquake, the overturning moments and shears due to the jet reaction load resulting from a severed pipe, and the compressive effects on the support skirt due to the thermal and pressure expansion of the reactor vessel. The expected maximum earthquake loads for the Hanford 2 reactor vessel support skirt are less than 50% of the maximum design basis loads used in the buckling analysis described; therefore, the expected faulted loads are well below the critical buckling limits of Paragraph F-1370(c) for this reactor vessel support skirt. The expected earthquake loads for this reactor were determined using the seismic dynamic analysis methods described in Section 3.7 of the WNP-2 Final Safety Analysis Report.

The assumption that the critical buckling stress in the material yield stress at temperature is not needed in the design analysis.

Summation: This item is closed.

DSER 3.9.3.4.b (MEB-43)

b. The applicant has supplied information concerning the design of the bolts and the baseplates as a response to our Office of Inspection and Enforcement Bulletin 79-02. The review of this information is being performed jointly by our Office of Inspection and Enforcement and Our Office of Nuclear Reactor Regulation. We will report the results of our review in a supplement to this Safety Evaluation Report.

Subject to resolution of the above open issues, our findings are as follows:

The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.4 Control Rod Drive Systems

Our review under Standard Review Plan Section 3.9.4 covered the design of the hydraulic control rod drive system up to its interface with the control rods. We reviewed the analyses and tests performed to assure the structural integrity and operability of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40 year life.

QUESTION NO. 43
(3.9.3.4)

The applicant has supplied information concerning the design of not only the bolts but also the baseplates into which the bolts are inserted and which the bolts connect to the underlying concrete or steel structures. This information has been submitted as a response to our Office of Inspection and Enforcement Bulletin 79-02, "Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts". The review of this information is being performed jointly by our Office of Inspection and Enforcement and our Office of Nuclear Reactor Regulation. We will report the results of our review in a supplement to this Safety Evaluation Report.

Summation - No action. Closed item for MEB.

b. The applicant has supplied information concerning the design of the bolts and the baseplates as a response to our Office of Inspection and Enforcement Bulletin 79-02. The review of this information is being performed jointly by our Office of Inspection and Enforcement and Our Office of Nuclear Reactor Regulation. We will report the results of our review in a supplement to this Safety Evaluation Report.

Subject to resolution of the above open issues, our findings are as follows:

The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or ether service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.4 Control Rod Drive Systems

Our review under Standard Review Plan Section 3.9.4 covered the design of the hydraulic control rod drive system up to its interface with the control rods. We reviewed the analyses and tests performed to assure the structural integrity and operability of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40 year life.

DSER 3.9.4 a (MEB-44)

The information presented in the FSAR, pertaining to the test programs which were conducted to verify the design, is inadequate to arrive at a conclusion as to whether the drives will function over the full range of temperatures, pressure, loadings and misalignments as required. Areas for which additional information is requested are:

a. Paragraph 3.9.4.3 (Page 3.9-73) states that deformation is not a limiting factor in the analysis of the CRD's components since the stresses are in the elastic region. This statement is not necessarily valid. It seems that elastic deformations and thermal deformations could possibly result in critical displacements. Have these areas been considered in the analysis?

b. Table 3.9-2(v) (Page 3.9-157) lists the stress limit for faulted conditions as: $S_{limit} = 1.2\ S_m = 1.2 \times 16660 = 20000$ psi., with a note: Analyzed to emergency conditions limits. Then in the column of Allowable Stress is listed 24990 psi., and a calculated stress of 22030. The calculated stress is within the limits for an allowable stress of 24990 but not for an allowable stress of 20000 psi. Clarification is requested of this area (Reference Section 3.9.3.1(a) of this Draft SER).

Subject to resolution of the above open issues, our findings are as follows:

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the control rod drive system are in conformance with Standard Review Plan Section 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27.

QUESTION NO. 44
(3.9.4)

Paragraph 3.9.4.3 (Page 3.9-73) states that deformation is not a limiting factor in the analysis of the CRD's components since the stresses are in the elastic region. This statement is not necessarily valid. It seems that elastic deformations and thermal deformations could possibly result in critical displacements. Have these areas been considered in the analysis?

RESPONSE

Elastic and thermal deformation have both been considered in the design of the reactor internals and control rod drives to ensure that the rod insertability is not affected, i.e. no mechanical interference, during and after an accident. Studies show that no plastic deformation occurs.

Summation - This item is closed.

PCY:rf/45E15
8/18/81

DSER 3.9.4b (MEB-45)

The information presented in the FSAR, pertaining to the test programs which were conducted to verify the design, is inadequate to arrive at a conclusion as to whether the drives will function over the full range of temperatures, pressure, loadings and misalignments as required. Areas for which additional information is requested are:

a. Paragraph 3.9.4.3 (Page 3.9-73) states that deformation is not a limiting factor in the analysis of the CRD's components since the stresses are in the elastic region. This statement is not necessarily valid. It seems that elastic deformations and thermal deformations could possibly result in critical displacements. Have these areas been considered in the analysis?

b. Table 3.9-2(v) (Page 3.9-157) lists the stress limit for faulted conditions as: $S_{limit} = 1.2\ S_m = 1.2 \times 16660 = 20000$ psi., with a note: Analyzed to emergency conditions limits. Then in the column of Allowable Stress is listed 24990 psi., and a calculated stress of 22030. The calculated stress is within the limits for an allowable stress of 24990 but not for an allowable stress of 20000 psi. Clarification is requested of this area (Reference Section 3.9.3.1(a) of this Draft SER).

Subject to resolution of the above open issues, our findings are as follows:

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the control rod drive system are in conformance with Standard Review Plan Section 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability stipulations of General Design Criterion 27.

QUESTION NO. 45
(3.9.4)

Table 3.9-2(v) (pages 3.9-167) lists the stress limit for faulted conditions as: S_{limit} = 1.2 Sm = 1.2 x 16660 = 20,000 psi, with a note: Analyzed to emergency conditions limits then in the column of Allowable Stress is listed 24990 psi, and a calculated stress of 22030. The calculated stress is within the limits for an allowable stress of 24990 but not for an allowable stress of 20000 psi. Clarification is requested of this area (Ref. Section 3.9.3.1(a) of this draft SER).

RESPONSE

At the time the allowable stress was originally calculated, the emergency limit for "membrane plus bending" was 1.5 Sm or 24,990 psi. Since then, the code has adopted an S'm which is 1.2 Sm. Therefore the allowable is now 1.5 S'm = 1.5 x (1.2 x Sm) = 1.5 x (1.2 x 16,660) = 29,990 psi. Accordingly, the table entry is revised as attached.

Summation - This item is closed.

PCY:ggt:rf/45L17
9/23/81

TABLE 3.9-2 (v)

CONTROL ROD DRIVE HOUSING

Operating Condition		Loading Combinations
A.	Normal & Upset	$P_D + F_{SR} + W + OBE$
B.	Emergency	$P_P + F_{SRP} + W + SSE$

Stress Limits:

The stress limits for the CRD are per ASME Boiler and Pressure Vessel Code and are listed on the attached tables.

P_D: Design pressure

P_P: Peak pressure

F_{SR}: Load due to stuck rod scram at design pressure

F_{SRP}: Load due to stuck rod scram at peak pressure

W: Static weights

A1

TABLE 3.9-7 (V) (Continued)

Criteria	Loading	Primary Stress Type	Allowable Stress (psi)	Calculated Stress (psi)
Primary Stress Limit - The allowable primary membrane stress is based on the ASME Boiler and Pressure Vessel Code, Section III, for Class I vessels, for type 304 stainless steel.	Normal and upset condition loads 1. Design pressure 2. Stuck rod scram loads 3. Operational basis earthquake, with housing lateral support installed.	Maximum membrane stress intensity occurs at the tube to tube weld near the center of the housing for normal, upset and emergency conditions.	16,660	11,900
			16,660	14,480 ~~[illegible]~~
For normal and upset conditions S_m = 16,660 psi @ 575°F				
For faulted conditions: ~~S_{limit} = 1.2S_m = 1.2 x 16,660 = 20,000 psi~~ Note: Analyzed to emergency conditions limits. $S_{limit} = 1.5 \times 1.2 \times S_m$ $= 29{,}990$ psi	Emergency conditions loads 1. Design pressure 2. Stuck rod scram loads 3. Design basis earthquake, with housing lateral support installed.		29,990 ~~21,990~~	22,030

DSER 3.9.5a (MEB-46)
DSER 3.9.5b (MEB-47
DSER 3.9.5c (MEB-48)

3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan (SRP) Section 3.9.5 is concerned with the load combinations, allowable stress limits, and other criteria used in the design of the WNP-2 reactor internals.

Our review has resulted in the following open issues.

a. Table 3.9-13 establishes stress intensity limits for the core support structure faulted loading conditions. As this table is somewhat different than the limits from Section III Appendix F, what is the basis and justification for Table 3.9-13? Would the computed stresses be in compliance with the faulted condition limits of Section III Appendix F?

b. It is the staff position that all BWRs under construction should document their actions being taken with respect to the problem of cracking of jet pump holddown beams. We will require the applicant's response to the letter from R. Tedesco to N. Strand, "Cracking of BWR Jet Pump Holddown Beam," dated August 5, 1980.

c. We will require the applicant to provide a commitment to NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking."

Subject to resolution of these issues, our findings are as follows:

The specified transients, design and service loadings, and combinations of loadings as applied to the design of the WNP-2 reactor internals provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these reactor internals would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss

QUESTION NO. 46
(3.9.5)

Table 3.9-13 establishes stress intensity limits for the core support structure faulted loading conditions. As this table is somewhat different than the limits from Section III Appendix F, what is the basis and justification for Table 3.9-13? Would the computed stresses be in compliance with the faulted condition limits of Section III Appendix F?

RESPONSE

The limits outlined in Table 3.9-13 were based on a draft of ASME Code Section III Subsection NG issued in January 1971. The limits are not significantly different from those shown in Appendix F of the current code. The attached Table shows that in many cases 3.9-13 is more conservative than Appendix F. But in one case it is slightly lower (0.75 Su instead of 0.7 Su). Overall there are no significant differences between the 2 sets of limits. It is therefore shows that the stresses would meet Appendix F also.

TABLE 3.9-13

CORE SUPPORT STRUCTURES

STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR FAULT CONDITIONS

3.6Sm Appendix F

0.7Su per Appendix F

Essentially same as Appendix F

STRESS CATEGORIES	PRIMARY STRESSES: MEMBRANE, P_m (NOTES 1,2&3)		PRIMARY STRESSES: BENDING, P_B (NOTES 1,2&3)		SECONDARY STRESSES: MEMBRANE & BENDING SECONDARY, Q	PEAK STRESSES: PEAK F
	P_m		$P_m + P_B$			
FAULT (NOTE 9)	$2.4S_m$	ELASTIC ANALYSIS	$3.0S_m$	ELASTIC ANALYSIS	EVALUATION NOT REQUIRED	EVALUATION NOT REQUIRED
	OR		OR			
	$0.75S_u$	(NOTE 5)	$1.33L_L$	LIMIT ANALYSIS (NOTE 4)		
	OR		OR			
	$1.33L_L$	LIMIT ANALYSIS (NOTE 4)	$0.75S_u$	PLASTIC ANALYSIS (NOTES 5&6)		
	OR		OR			
	$0.67S_u$	PLASTIC ANALYSIS (NOTES 5&6)	$0.8L_T$	TEST (NOTE 7)		
	OR		OR			
	$0.8L_T$	TEST (NOTE 7)	KS_T	STRESS-RATIO ANALYSIS (NOTE 8)		
	OR					
	S_T	STRESS-RATIO ANALYSIS (NOTE 8)				

TABLE 3.9-12 (Continued)

the ultimate strength or other governing material properties of the actual part and the tested parts to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for emergency conditions.

NOTE 8 - Stress ratio is a method of plastic analysis which uses the stress ratio combinations (combination of stresses that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. K is defined as the section factor; $S_e \leq 2S_m$ for primary membrane loading.

NOTE 9 - Where deformation is of concern in a component, the deformation shall be limited to two-thirds the value given for Emergency Conditions in the Design Specification.

NOTE 10 - When loads are transiently applied, consideration should be given to the use of dynamic load amplification and possible change in modulus of elasticity.

QUESTION No. 47

It is the staff position that all BWR's under construction should document their actions being taken with respect to the problem of cracking of jet pump holddown beams. We will require the applicant's response to the letter from R. Tedesco to N. Strand, "Cracking of BWR Jet Pump Holddown Beam", dated August 5, 1980.

RESPONSE

The supply System's response to the letter from R. Tedesco to N. Strand "Cracking of BWR Jet Pump Holddown Beam", dated August 5, 1980, is contained in the letter from G. Bouchey to R. Tedesco dated December 4, 1980 (GO2-80-279). This letter states the action which will be taken by the Supply System with respect to the problem of jet pump holddown beam cracking.

Summation - This item is closed.

THIS LETTER SATISFIES COMMITMENT NO. ____________

THIS LETTER (DOES) (DOES NOT) ESTABLISH A NEW COMMITMENT

WPPSS CORRESPONDENCE NO. ____________

INTERNAL DISTRIBUTION

GD Bouchey--050
WG Conn--906D
OK Earle--906D
CL Fies--065
BA Holmberg--901A
JR Lewis (BPA)--505
GC Sorensen--420
IEB File
~~WNP-2 Files~~

LT Harrold--410
ELE-KAH-80-067
cc-906D

December 4, 1980
G02-80-279

Docket No. 50-397

Mr. R. L. Tedesco, Assistant Director, Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Tedesco:

Subject: CRACKING OF BWR JET PUMP HOLDDOWN BEAMS

Ref.: Letter, R. L. Tedesco to N. O. Strand, same subject, dated August 5, 1980

FILE COPY

The referenced letter requested the Supply System provide specific information regarding actions taken to preclude cracking of jet pump holddown beams. The following responses correspond directly with the questions posed in your letter.

1. WNP-2 jet pump beams have been installed, but will be retensioned from a 30 kip preload to a 25 kip preload before fuel load. This is expected to increase beam operating time to crack initiation at the 2.5% probability level to a range of 19 to 40 years

   During operation, periodic inspections will be conducted as part of our overall inservice inspection program. Inspection frequencies will be developed in the future based on lead plant inspection results and the results of future GE testing. These inspections should provide adequate warning of potential beam failure.

2. It is our position that reducing the tension preload to 25 kips on the beams provides an adequate long term solution. If a problem is still present, as identified by our inservice inspections, improved heat treated beams may be purchased from GE. Tests indicate the improved beams may provide double the time to crack initiation as compared to the current beams.

no letterhead rec'd

AUTHOR	KA Hadley		FOR SIGNATURE OF:	GD Bouchey
SECTION				
FOR APPROVAL OF	GC Sorensen	LT Harrold	DL Renberger	
APPROVED	[signature]	LT Harrold	[signature]	
DATE	11/20/80	11/25/80	[illegible]	

WP 114