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AUTH.NAME	AUTHOR AFFILIATION
RENBERGER,D.L.	WA PUB PWR SUPPLY SYS

RECIP.NAME	RECIPIENT AFFILIATION
VARGA,S.A.	***LIGHT WATER REACTORS BRANCH 4

SUBJECT: Forwards copies of responses to NRC 780918 ltr requesting addl info re Instrumentation & Controls Sys.Responses to be formally inserted in FSAR as an Amend in Feb 1979. (see reports)

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NOTES: BILL PATON (OELD) 1CY ER AMDTS.

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	06 I & E	2	2	08 OPERA LIC BR	1	1
	09 GEOSCIEN BR	1	1	10 QAB	1	1
	11 MECH ENG BR	1	1	12 STRUC ENG BR	1	1
	13 MATL ENG BR	2	2	15 REAC SYS BR	1	1
	16 ANALYSIS BR	1	1	17 CORE PERF BR	1	1
	18 AUX SYS BR	1	1	19 CONTAIN SYS	1	1
	20 I & C SYS BR	1	1	21 POWER SYS BR	1	1
	22 AD SITE TECH	4	4	26 ACCDNT ANLYS	1	1
	27 EFFL TRT SYS	1	1	28 RAD ASMT BR	1	1
	29 KIRKWOOD	1	1	AD FOR ENG	1	0
	AD PLANT SYS	1	0	AD REAC SAFETY	1	0
	AD SITE ANLYSIS	1	0	DIRECTOR NRR	1	0
	MPA	1	0	OELD	1	0
				HYDRO-METEOR BR W/ENCL		
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	30 ACRS	16	16			

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Washington Public Power Supply System
A JOINT OPERATING AGENCY

P. O. Box 968 3000 Geo. Washington Way Richland, Washington 99352 Phone (509) 375-5000

G02-79-04
January 5, 1979

Docket No. 50-397

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

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

Subject: WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO INSTRUMENTATION AND CONTROL BRANCH, REACTOR SYSTEMS BRANCH, AND HYDROLOGY-METEOROLOGY BRANCH QUESTIONS

Reference: (1) Letter, SA Varga, NRC to WPPSS, "Additional Acceptance Review Questions for the WPPSS Nuclear Plant No. 2", dated Sept. 18, 1978

(2) Letter, WPPSS to NRC, "Responses to Containment Systems Branch, Core Performance Branch, and Geoscience Branch Questions", dated Nov. 21, 1978.

Dear Mr. Varga:

Attached please find sixty (60) copies of responses to the reference 1 questions from the subject branches. This submittal, in conjunction with the responses submitted with reference 2, substantially completes action on the reference 1 question set and is consistent with the schedule provided you at the October 10th meeting. The few open items

B001 SE 1/60
ADD HYDRO-METEOR BR W/ENCL

remaining from the question set are being carried forward and will be submitted at the earliest possible date: These responses will be formally inserted in the FSAR in an amendment in February.

Very truly yours,

D L Renberger

D. L. RENBERGER
Assistant Director
Technology

DLR:OKE:cph

cc: JJ Verderber, B&R w/o responses
JJ Byrnes, B&R "
RC Root, B&R Site "
HR Canter, B&R "
D. Roe, BPA "
FA MacLean, GE San Jose "
I. Littman, WPPSS, NY "

E. Chang, GE San Jose w/5 copies of responses
J. Ellwanger, B&R " "

NS Reynolds, Debevoise & Liberman w/1 copy of responses
WNP-2 Files " "

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

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

DATED Jan. 4, 1979

D L Renberger
D. L. RENBERGER

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

GIVEN under my hand and seal this 4 day of January, 1979.

[signature]
Notary Public in and for the State
of Washington
Residing at Kennewick

Instrumentation

and

Controls Systems

Branch Responses

Q. 031.055
(3.10.1.1)
(031.001)
(031.044)

Identify all class 1E equipment which was not qualified by test. For each such item, provide the basis for assuming that it will not be spuriously operated, or fail to operate when required, during and after a seismic event.

RESPONSE:

An extensive seismic and environmental review program is presently underway encompassing BOP and NSSS scope, with a planned completion date in the second quarter of 1979.

Within the BOP scope, the equipment documentation has been extracted from the contract files, copied and categorized for easy retrieval. Within the NSSS scope, contract negotiations are underway with GE to perform a similar function.

A list of all Class 1E equipment including splices, terminal blocks, termination cabinets and connectors is presently being compiled. This list will contain the following information:

1. Equipment location
2. Safety functional requirement
3. Manufacturer & Model No.
4. Qualification Method (test-analysis)
5. Environmental Extremes
6. Identification and location of qualification documents

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

The completed list will be included in the FSAR as equipment tables in sections 3.10 (seismic) and 3.11 (environmental).

The extensive review program underway will also satisfy the requirements of IE circular 78-08.

Q. 031.056
(3.11)

Describe the environmental qualification procedures and the environmental extremes of qualification for the following specific passive Class 1E components inside the dry well: (1) splices; (2) terminal blocks; (3) termination cabinets; and (4) connectors.

RESPONSE:

See response to Q. 031.55.

Q. 031.057
(3.11)

Identify all class 1E equipment inside the drywell, except the equipment cited in Item 031.056, and summarize the environmental qualifications for this equipment. The identification and summary for each item should include: (1) the safety function and functional requirement, (2) the manufacturer, model number, type number, and any other identifying numbers; (3) the specific location of this equipment in the drywell; (4) the method of environmental qualification; (5) the environmental extremes, including the time period of testing, for which it is qualified; and (6) the identification and the location of the documents which are available so as to permit an independent evaluation of the adequacy of the environmental qualification.

RESPONSE:

See response to Q. 031.55.

Q. 031.058
(031.001)

Your response to Item 031.001(e) is incomplete. Describe the passive design features which prevent motion of the recirculation flow control valve after an accident. This motion could result from postulated damage to the hydraulic system during a loss of coolant accident.

RESPONSE:

As stated in the response to 031.001(e), the valve is considered inactive since, if it moves at all, it will move in the open direction in a LOCA.

The passive design features associated with the recirculation flow control valve to resist unaccountable motion in an accident are as follows:

1. Minimum exposure of components to the accident environment - The only components in the recirculation flow control valve system subject to the direct effects of a LOCA are the valve, valve actuator , and hydraulic lines. The balance of the system I and C is located outside primary containment.

2. Physical remoteness from potential pipe breaks -

   a) Referring to the response to 31.001(e), it is recalled the valve of concern is the one in the unbroken loop. Accordingly, for recirc. pipe breaks, there is virtually no potential for missile or jet impingement impact on the valve in the unbroken loop since it is on the opposite side of the reactor vessel within primary containment from the other loop.

   b) For other than recirc. pipe breaks, over 25 vertical feet of separation for the actuator and 15 vertical feet of separation for the hydraulic lines exist from the nearest high energy line.

3. Physical Protection from Effects of Pipe Breaks
   Substantial direct physical protection is offered as follows:

   a) Valve actuator - The valve actuator is located within the confines of large structural beams at the 512 foot elevation.

   b) Hydraulic lines - The hydraulic lines are enclosed in schedule 40 pipe for protection purposes from the containment penetration to just above the valve actuator.

[illegible]

Accordingly, the risk is minimal for direct LOCA effects and the valve can be expected to function as described in the response to 031.001(e). However, in the extremely unlikely event there is a direct impact, system response is still satisfactory as follows:

1. <u>Loss of hydraulic lines</u> -

   The hydraulic actuator has two integrally mounted pilot operated hydraulic check valves in each of the drive lines which prevent significant control valve movement should any of the three hydraulic lines break. If the hydraulic supply line to the pilot actuated valves were to break, the check valves would immediately close and the recirculation flow control valve actuator would lock in the as-is position. If either of the two or both of the driving lines were to break, a number of system interlocks would stop the hydraulic pump and the pilot operated check valves once again would close, thus locking the actuator in position. In this case, the differential pressures across the hydraulic cylinder could cause a slight control valve movement of 1 to 3 percent of total valve movement before being locked up.

2. <u>Loss of Power</u> -

   On loss of control power, motive power, or hydraulic power, the actuator will be disabled and the valve will lock in the as-is position.

A complete system failure mode and effects analysis along with a more detailed system description are described in the soon to be submitted "Appendix H" of the FSAR. (Scheduled to be submitted in the first quarter of 1979).

[illegible]

Q. 031.059 (031.001) (11.6-1)

Your response to Item 031.001(h) presents a new design for the logic of the main steam line isolation valves which is different from that reviewed and accepted for licensing on similar boiling water reactors. Provide the manufacturing drawings for ASCO Valve No. 832320. Additionally, provide the results of the engineering analysis and the test results which demonstrate the ASCO Valve No. 832320: (1) is qualified for the environment in the drywell following a loss of coolant accidnet; (2) is seismically qualified; (3) meets the physical separation and the required electrical independence in accordance with the staff positions contained Regulatory Guide 1.75; (4) satisfies the single failure criterion (previous designs accepted for licensing have used two separate valves in a one-out-of-two logic for a reactor trip). Note that Table 1.6-1 of the FSAR states that the GE Topical Report, APED-5750, is applicable to the WNP-2 facility and that Table 7.1-1 indicates the main steam line isolation valves are designed and supplied by GE. Accordingly, provide justification for the change to the design which was previously reviewed and approved by the staff in our evaluation of the GE Topical Report, APED-5750.

RESPONSE:

The main steam line isolation valve logic is the same for WNP-2 as that supplied for previously reviewed and accepted for licensing BWR's.

Attachment - ASCO DWG. #HVA-166-265

(1) ASCO Valve #832320 is a fail safe valve and closes by current deenergizing. This valve's application does not require it to function following a LOCA. The valve has been qualified for normal ambient conditions (VPF #3680-1) as follows:

Cycle tested at temperature (172-198^{o}F)

100 cycles at 4 minute intervals
10 cycles at 12 hours intervals
5 cycles at 120 hours intervals

Total time at temperature 941 hours

(2) Qualification of valve for seismic. The solenoid valve was qualified for original seismic requirement when tested with complete valve (Wyle Laboratories -- Seismic Simulation Test Report #42610-1, dated 2/27/74).

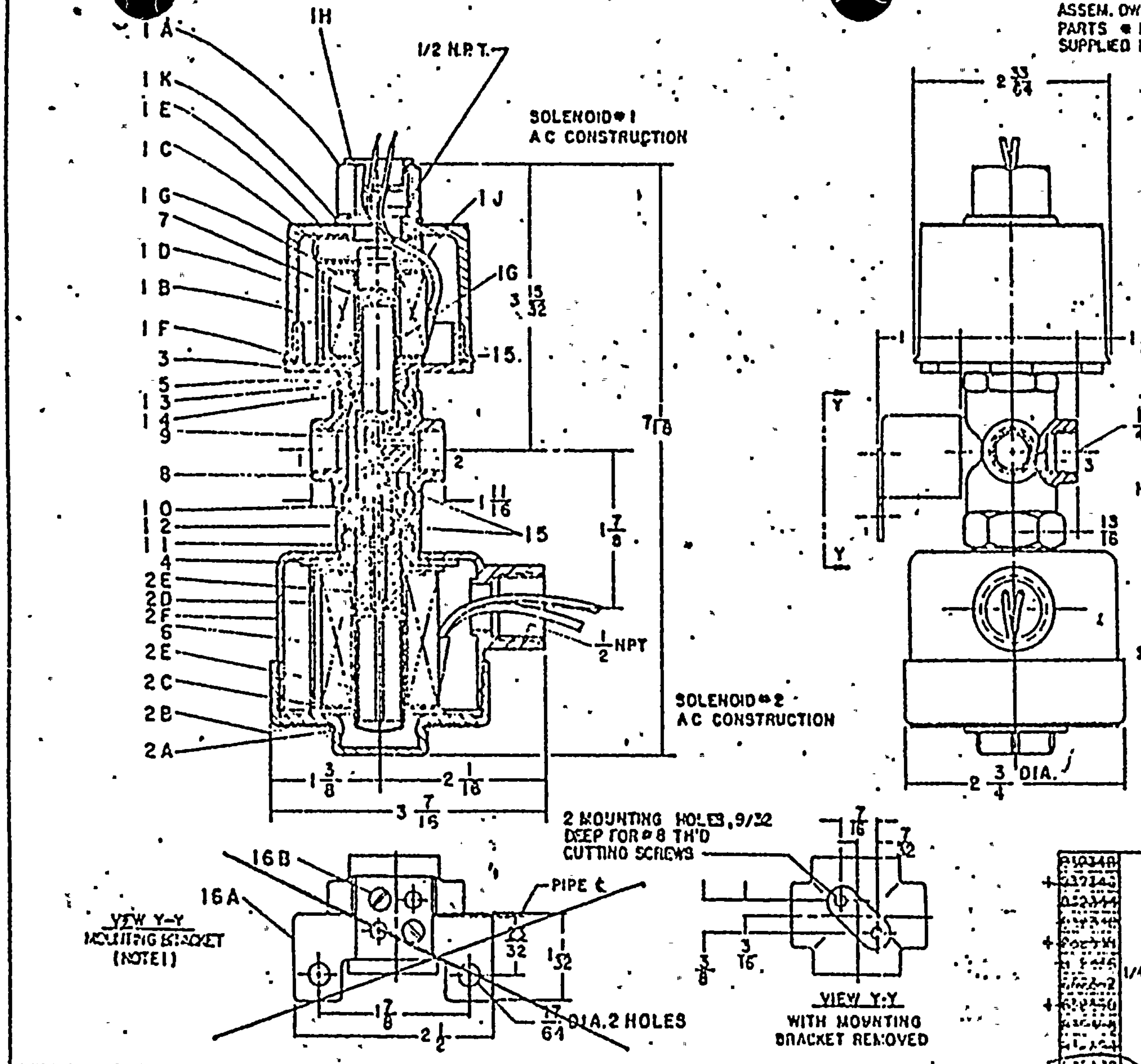
The solenoid valve remained funtional during all phases of the testing.

(3) The protection system criteria of IEEE 279-1971 are met with this design; the requirements of Reg. Guide 1.75 were not committed for this plant.

(4) The ASCO valves in question are not used in generating a reactor trip. The ASCO valves are used in a two out of two logic for each MSIV. That is, in order for each MSIV to be isolated both ASCO solenoids must deenergize. The ASCO valves themselves are not single failure proof. Single failure criterion is preserved since each main steam line contains two valves in series. If a single failure occurs in one valve scheme the second will provide isolation.

There is no deviation from the commitments made in APED-5750.

GIVE FULL NAMEPLATE DATA, AND ASSEM. DWG NO. WHEN ORDERING PARTS * INDICATES ITEMS SUPPLIED IN SPARE PARTS KITS.



NOTES:

1- OPTIONAL MOUNTING BRACKET SUPPLIED ONLY WHEN SPECIFIED "WITH BRACKET". BRACKET MAY BE ROTATED TO ANY OF FOUR POSITIONS, IN INCREMENTS OF 90° FROM POSITION SHOWN.

2- + USE CLASS 'B' HIGH POWER COIL FOR CATALOG NO'S. 832322, 832330, 832330 & 832346 ONLY. (SOL. # 1)

PARTS LIST

ITEM	NAME	QTY
1 A	HOUSING	1
1 B	YOKE	1
1 C	SPACER	1
1 D	COIL	1
1 E	NAMEPLATE	1
* 1 F	GASKET, HOUSING	1
1 G	SLEEVE	2
1 H	THREAD PROTECTOR	1
1 J	TUBING-INSULATING	1
1 K	SNAP RING	1
2 A	RETAINING RING	1
2 B	NAMEPLATE	1
2 C	YOKE	1
2 D	COIL	1
2 E	SLEEVE (A C)	2
2 F	COVER/HOUSING SUB-ASSEMBLY	1
3	SOL BASE SUB-ASSEM. (UPPER)	1
4	SOL. BASE SUB-ASSEM. (LOWER)	1
* 5	CORE ASSEM. (UPPER)	1
* 6	CORE ASSEM. (LOWER)	1
* 7	SPRING, CORE	1
8	BODY	1
* 9	DISC HOLDER ASSEM	1
* 1 0	SPRING, DISC	1
1 1	STEM	1
1 2	PLUGNUT ADAPTER ASSEMBLY	1
1 3	ADAPTER	1
* 1 4	GASKET, ADAPTER	1
* 1 5	GASKET, BODY	3
1 6 A	BRACKET, MOUNTING	1
1 6 B	SCREW, BRACKET	2

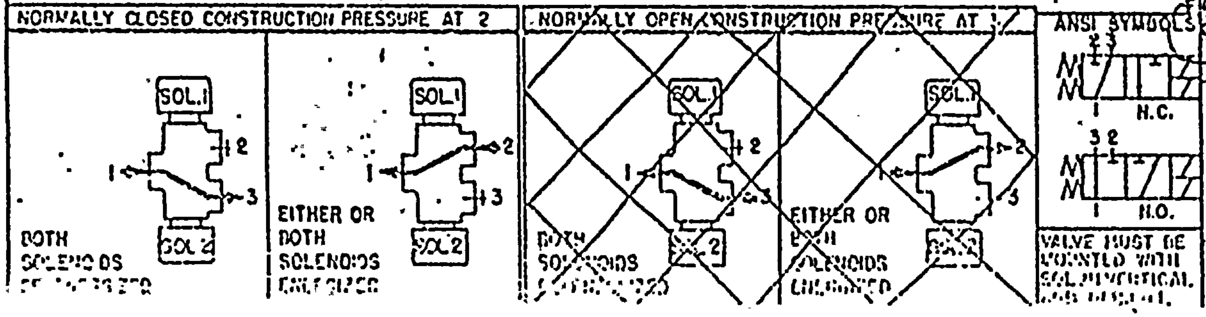
CATA. NO.	PIPE	ORIF.	OPER.	WATTAGE SOL. # 1	WATTAGE SOL. # 2	SPK. NUMBER	NET WEIGHT
832348	1/4	7/8	NORM. OPEN	6 A C	10.0 A.C.	FV160-263	3.1 LB.
+ 832342		3/32		9 A C		FV160-260	
[illegible]		1/16		6 A C		FV166-265	
[illegible]		[illegible]	NORM. CLOSED				
+ [illegible]		[illegible]		9 A C		FV166-262	
[illegible]		[illegible]		6 A C	15.4 A.C.	FV160-263	
[illegible]		7/8	NORM. OPEN				
+ [illegible]		[illegible]		9 A C		FV160-262	
[illegible]		[illegible]		6 A C		FV160-265	
[illegible]		[illegible]	NORM. CLOSED	9 A C		FV-166-262	
832320		1/16		6 A C			

ANSI SYMBOLS

N.C.

N.O.

VALVE MUST BE MOUNTED WITH SOL. IN VERTICAL [illegible]



BULLETIN 8323 3 WAY

DUAL SOLENOID VALVE

PIPE CONN. 1/4 N.P.T. BODY MAT'L BRASS

ORIFICE DIA. [illegible] SEATING RESILIENT

ENCLOSURE [illegible] WATERTIGHT & EXP. PROOF

WATTAGE (SEE TABLE)

HVA-166-265

PAGE 12

INSTALLATION AND MAINTENANCE INSTRUCTIONS

GENERAL PURPOSE AND EXPLOSION PROOF WATERTIGHT SOLENOIDS

CATALOG
80161 80163
80162 80164

DESCRIPTION

Catalogs 80161 and 80162 solenoids are equipped with a General Purpose, NEMA Type 1 Enclosure. When Catalog 80161 or 80162 is installed just as a solenoid and not attached to an ASCO valve, the core has a 1/4-28-UNF-2B tapped hole, 3/8 full thread.

Catalogs 80163 and 80164 solenoids are equipped with an enclosure which is designed to meet NEMA Type 4 — Watertight, NEMA Type 7 (C or D) Hazardous Locations — Class 1, Group C or D and NEMA Type 9 (E, F or G) Hazardous Locations — Class II, Group E, F or G. When Catalog 80163 or 80164 is installed just as a solenoid and not attached to an ASCO valve, the core has a 1/4-28-UNF-2B tapped hole, 3/8 full thread.

OPERATION

When the solenoid is energized, the core pulls into the solenoid base sub-assembly. IMPORTANT: When the solenoid is de-energized the initial return force for the core whether developed by spring, pressure or weight must exert a minimum force to overcome residual magnetism created by the solenoid. Minimum return force for A-C Construction is 11 ounces and 5 ounces for D-C Construction.

INSTALLATION

Check nameplate for correct catalog number, voltage, wattage and service.

IMPORTANT: For the protection of the solenoid valve or operator install a strainer or filter suitable for the service involved in the inlet side as close to the valve or operator as possible. Periodic cleaning is required depending on service conditions. See Bulletins 8600, 8601 and 8602 for strainers.

WIRING

Wiring must comply with Local and National Electrical Codes. For Catalog Numbers 80163, 80164 electrical fittings must be approved for use in the approved hazardous locations. Housings for all solenoids are made with connections for 1/2 inch conduit. The general purpose enclosure may be rotated to facilitate wiring by removing the retaining cap. Rotate enclosure to desired position. Replace retaining cap before operating.

NOTE

Alternating Current (AC) and Direct Current (DC) solenoids are built differently. To convert from one to another, it is necessary to change the complete solenoid.

CAUTION

Care must be taken not to mar the core surface.

SOLENOID ENCLOSURE ASSEMBLY

Catalog 80161, 80162, 80163, and 80164 may be assembled as a complete unit. Tightening is accomplished by means of a hex flange at the base of the solenoid enclosure.

SOLENOID TEMPERATURE

Standard solenoids are supplied with coils designed for continuous duty service. When the solenoid is energized for a long period, the solenoid enclosure becomes hot and can be touched by hand only for an instant. This is a safe operating temperature. Any excessive heating will be indicated by the smoke and odor of burning coil insulation.

MAINTENANCE

CLEANING

A periodic cleaning of all solenoid valves and operators is desirable. The time between cleanings will vary, depending on the media and service conditions. In general, if the voltage to the coil is correct, sluggish valve operation, excessive leakage or noise will indicate that cleaning is required.

IMPROPER OPERATION

1. Check the electrical system by energizing the solenoid. A metallic click signifies the solenoid is operating. Absence of the click indicates loss of power supply. Check for loose or burned-out fuses, open-circuited or grounded coil, broken wires or splice connections. Replace coil if necessary.
2. Check voltage across the coil leads with a voltmeter. Voltage must be at least 85% of nameplate rating.

COIL REPLACEMENT

1. Turn off electrical power and disconnect coil lead wires.
2. Remove parts in the following manner: (Refer to exploded view on reverse side.
   A. Catalog 80161, 80162
      1. Remove retaining cap, nameplate and housing cover.
      2. Slip yoke containing coil, sleeves and insulating washers off the solenoid base sub-assembly. Insulating washers are omitted when a molded coil is used.
      3. Coil is now accessible for replacement. Reassemble in reverse order of disassembly.
   B. Catalog 80163, 80164
      1. Unscrew housing cover with retaining ring and nameplate attached.
      2. Remove spacer from the top of the yoke.
      3. Slip yoke containing coil, sleeves and insulating washers off the solenoid base sub-assembly. Insulating washers are omitted when a molded coil is used.
      4. Coil is now accessible for replacement. Reassemble in reverse order of disassembly.

CAUTION

The solenoid must be fully reassembled as the housing and internal solenoid parts are part of and complete the magnetic circuit.

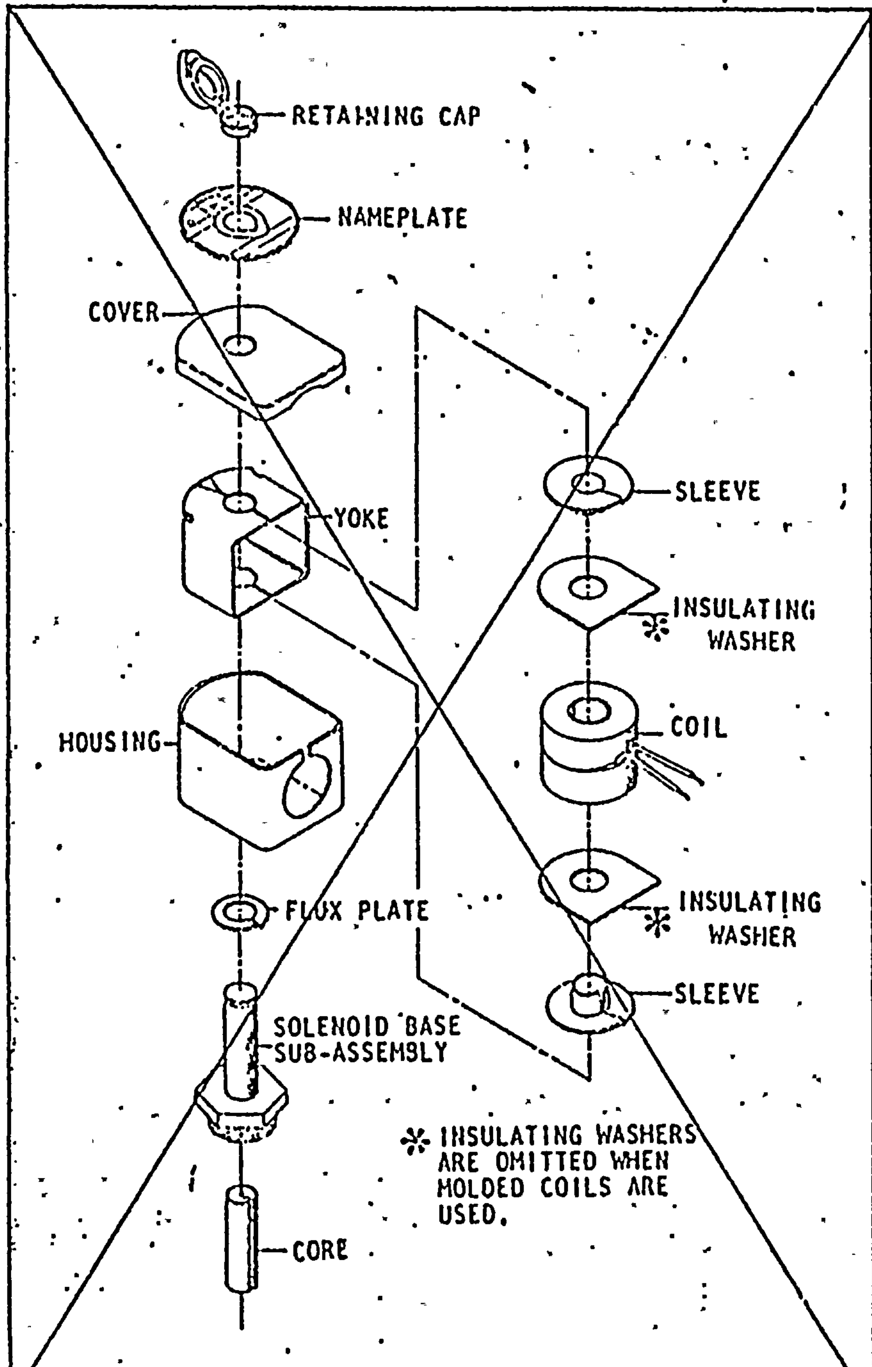
NOTE:

Catalog 80163, 80164

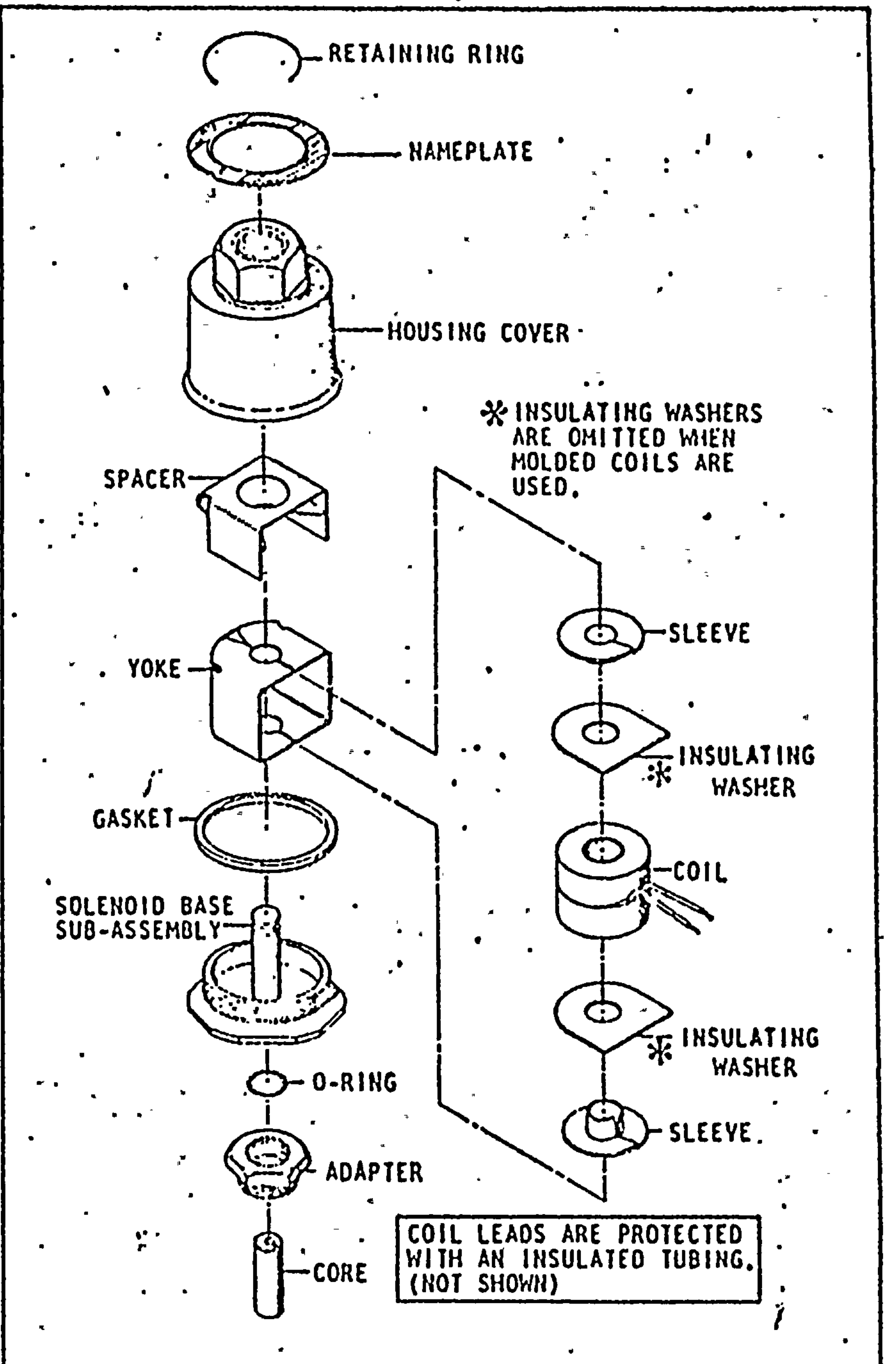
Installation and Maintenance of Explosion Proof equipment requires more than ordinary care to insure safe performance. All finished surfaces of the solenoid are constructed to provide a flameproof seal. Be sure that surfaces are wiped clean before replacing.

ORDERING INFORMATION FOR SPARE PARTS
FOR SPARE PARTS SPECIFY SERIAL NUMBER, CATALOG NUMBER, VOLTAGE AND INDIVIDUAL PARTS BY NAME.

FORM NO V6391R1 PRINTED IN U.S.A. 1971 ASCO Valves Automatic Switch Co. FLORHAM PARK, NEW JERSEY



CATALOG 80161, 80162



CATALOG 80163, 80164

SCN 78-52

Q. 031.060
(RSP)
(031.001)

Your response to Item 031.001(i) is unacceptable since it contradicts information contained in Reference 2 of your response. Specifically, the short term objectives presented in parts 1 and 2 of your response appear to preclude prompt cessation of feedwater flow which is stated in part 3 of the response to be desireable. Accordingly, we require you to provide a response which is consistent with the material which has been submitted by General Electric to the staff for our review of anticipated transient without scram.

RESPONSE:

An inconsistency does not exist because parts 1 and 2 of Question 031.001(i) should not be compared with part 3. The coast down of feedwater flow as in the ATWS studies mitigates the vessel pressure, containment temperature and pressure rises and, as such, is a benefit in the initial stages of the transient. By the time the feedwater lines are needed for makeup water, the transient (as far as ATWS is concerned) is over. Using the feedwater lines to supply makeup water from the condensate system pumps is considered for "short term" containment isolation but, in fact, occurs chronologically much later than in the ATWS studies.

Q. 031.061
(031.039)

Your response to Item 031.039(b) is incomplete. Describe the design provisions which eliminate the use of jumpers other than test instrumentation leads.

RESPONSE:
(031.061)

Jumpers are not used except for the SCRAM initiation on SRM high flux trip during initial plant startup. This trip is jumpered out with permanent terminal block links after startup is completed.

Q. 031.062

It is the staff's position that the responses to Items 031.001(ff), 031.002, and 031.023 do not provide the detailed design information which is required for an independent review of an application for an operating license. Accordingly, we require you to provide amended responses to the items cited above. Specifically, identify the bus requested in Item 031.001(ff) provide the information requested in Item 031.002(e) and Item 031.023(d).

RESPONSE: (to 31.001(ff) only)

The reactor manual control system and refueling interlocks are supplied at 120VAC via the 120/240 volt critical (Class 1E) instrumentation ac power system panels PP-7A and PP-8A, except for the fan control alternate source which is supplied via the 120/240 volt plant critical ac power system panel PP-8A-A. Details of supply are as follows:

Panel	Div.	G.E. Board	Via
PP-8A	2	H13-P603	PP-8A-Z, H12-P679
PP-8A-A	2	H13-P603	H13-P679
PP-7A	1	H13-P615	PP-7A-Z, H13-P688
PP-7A	1	H13-P616	H13-P688

RESPONSE: (to 31.002(e))

Drawings are provided in the response to question 031.068.

RESPONSE: (to 031.023(d))

See the response to question 031.055.

Q. 031.063
(031.006)
(031.014)
(031.016)
(031.026)
(031.047)

Your responses to Items 031.006, 031.014, 031.016, 031.026, and 031.047, are unacceptable. Specifically, your responses to these particular items indicate that you will respond at a later date. Accordingly, provide a schedule for your response on these matters.

RESPONSES:

31.006 - See revised response.*

31.014 - See revised response.*

31.016 - Aditional information with reference to question 31.006 is as follows:

The tables referenced in the responses to question 31.016 and 31.021 are being revised as per the attached to indicate that the Technical Specifications for WNP-2 will contain instrumentation setpoint, response time, accuracy, and testing requirements necessary to insure minimum performance not less than those assumed in the safety analysis. Requirements listed currently in the tables are subject to change pending final approval of the Technical Specifications specific to WNP-2. The Technical Specifications and Instrumentation setpoints and bases for WNP-2 are based on NUREG 0123 Rev. 1, Standard Technical Specifications for General Electric Boiling Water Reactors, (April, 1978), and are in review at WPPSS having received the majority of the required input from General Electric. It is anticipated they will be ready for submission to the NRC in the second quarter of 1979.**

31.026 - See revised response.*

31.047 - See revised response.*

*See attached draft.

**See attached draft revised tables.

Revised Tables for

31.016

TABLE 7.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SPECIFICATIONS

SCRAM Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (8)	Required Accuracy (9)	Response (7) (9) Time
Reactor Vessel High Pressure	Pressure Switch	50-1200 psig	1050 psig	-	± 10 psi	1 sec.
Primary Containment High Pressure	Pressure Switch	0.2-6 psig	2.0 psig	-	± 0.05 psi	0.6 sec.
Reactor Vessel Low Water Level 3	Level Switch	0-60" (1)	+12.5" (1)	-	± 3"	—
Scram discharge Volume High Water Level	Level Switch	(5)	0" (4)	-	± .5"	—
Turbine Stop Valve Closure	Position Switch	0-100% open	10% Valve Closure	-	± 1%	≤10 mill sec.
Turbine Governor Valve Fast Closure	Pressure Switch	250-3000 psig	850 psig	-	± 60 psi	≤30 mill sec.
Main Steam Line Isolation Valve Closure	Position Switch	0-100% open	10% Valve Closure	-	± 1%	—
Neutron Monitoring System	See Subsection 7.6.1.1.5					
Main Steam Line High Radiation	Gamma Detector	$1-10^6$ cpm	3X normal	-	±30,000 cpm	1 sec.
Bypass Function						
Discharge Volume High Water Level Trip Bypass	N/A	N/A	Bypass switch in Bypass and Mode Switch in shutdown or refuel	-	N/A	N/A
Turbine Stop Valve and Governor Valve Fast Closure Trip Bypass	Pressure Switch	25-600 psig	Turbine first stage pressure < 30% Power (235 psig)	-	±20 psi	1 sec.

TABLE 7.2-1 (Continued)

SCRAM Function	Instrument	Instrument Range (2)	Trip Setting(3)	Margin(8)	Required Accuracy	Response(7) Time
Main Steam Line Isolation Valve Closure Trip Bypass	Pressure Switch	50-1200 psig	Reactor Pressure below 1045 psig and Mode Switch in Refuel, Startup or Shutdown	-	±15 psi at	1 sec.

NOTES FOR TABLE 7.2-1

(1) Instrument zero equal to 527.5" above Vessel Zero.

(2) See Chapter 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary. The initial values are the trip settings listed in the Tables.

(4) 0" equals 36 gal.

(5) Instrument range dependent on installation.

(6) ~~Response time will be supplied later.~~ Deleted.

(7) Output expressed as a function of time, resulting from application of a specified input under specified operation condition.

(8) Margins will be provided as part of technical specifications.

(9) Required values ~~[illegible]~~ will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.3-1

HIGH PRESSURE CORE SPRAY SYSTEM INSTRUMENTATION SPECIFICATIONS

HPCS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (8)	Required Accuracy (9)	Response Time (9)
Reactor Vessel Low Water Level (energize HPCS)	Level Switch	-150/0/+60" (1)	-38"	-	±7.5"	—
Primary Containment High Pressure	Presure Switch	.25-12 psig	2 psig	-	±0.06 psi	0.6 sec.
Reactor Vessel High Water Level Trip	Level Switch	0-60"	+55.5"	-	±3"	—
Pump Discharge Pressure	Pressure Switch	10-340 psig	120 psig	-	±10 psi	—
Pump Minimum Flow	Flow Switch	0-1190 gpm	640 gpm	-	±160 gpm	250 milli sec
Suppression Pool High Water Level	Level Switch	(6)	0" (4)	-	±0.5"	—
Condensate Storage Tank Low Level	Level Switch	(6)	0" (5)	-	±0.5"	—
Diesel Fuel Day Tank Level	Level Switch (H) Level Switch (L)	Float Type Float Type	El. 448'-2" El. 445'-0"	-	±0.5" ±0.5"	- -

NOTES FOR TABLE 7.3-1 Page 2 of 2

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chap. 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) 0" equal to 5" above normal water level.

(5) 0" equal to 11,500 gal.

(6) Instrument range dependent on installation.

(7) ~~Response time will be supplied later~~. Deleted.

(8) ~~Information to be supplied later.~~ Margins will be provided as part of the technical specifications.

(9) Required values ~~response times~~ will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.3-2

AUTOMATIC DEPRESSURIZATION SYSTEM INSTRUMENTATION SPECIFICATIONS

ADS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (6)	Required Accuracy (7)	Response Time (5) (7)
Reactor Vessel Low Water Level	Level Switch	-150/0/+60" (1)	-149"	-	±7.5"	—
Primary Containment High Pressure	Pressure Switch	0.25-12 psig	2 psig	-	±0.06 psi	0.6 sec.
LPCI Permissive	Pressure Switch	10-240 psig	100 psig	-	±9 psi	—
LPCS Permissive	Pressure Switch	10-340 psig	150 psig	-	±10 psi	100 milli sec.
Automatic Depressurization Time Delay	Timer	0-180 sec.	120 sec.	-	±18 sec.	—

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NOTES FOR TABLE 7.3-2

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) ~~Response time will be supplied later.~~ Deleted.

(5) Response time of instruments is not critical due to system delay of 120 sec.

(6) ~~Information to be supplied later.~~
Margins will be provided as part of the technical specifications.

(7) Required values will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.3-3

LOW PRESSURE CORE SPRAY INSTRUMENTATION SPECIFICATIONS

LPCS FUNCTION	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (5)	Required Accuracy (6)	Response Time (6)
Reactor Vessel Low Water Level (LPCS Initiation)	Level Switch	-150/0/+60" (1)	-149"	-	±7.5"	—
Primay Containment High Pressure (LPCS Initiation)	Pressure Switch	.25-12 psig	2 psig	-	±0.06 psi	0.6 sec
Injection Valve Differential Pressure	Differential Pressure Switch	0-800 psid	747 psid	-	±12 psi	100 milli sec.
Pump Minimum Flow Bypass	Flow Switch	0-1100 gpm	640 gpm	-	±170 gpm	250 milli sec.
Pump Discharge Pressure (signal to ADS)	Pressure Switch	10-340 psig	150 psig	-	±10 psi	—

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NOTES FOR TABLE 7.3-3 · Page 2 of 2

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) ~~Response time will be supplied later.~~ Deleted.

(5) ~~Information to be supplied later.~~
Margins will be provided as part of the technical specifications.

(6) Required values will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.3-4

LOW PRESSURE COOLANT INJECTION INSTRUMENTATION SPECIFICATIONS

LPCI Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (6)	Required Accuracy (7)	Response Time (7)
Reactor Vessel Low Water Level (LPCI Initiation)	Level Switch	-150/0/+60" (1)	-149"	-	±7.5"	—
Primary Containment High Pressure (LPCI Initiation)	Pressure Switch	.25-12 psig	2 psig	-	±0.06 psi	0.6 sec.
LPCI Pump Delay (on Loss of Normal Power)	Timer	0-7.5 sec.	5 sec.	-	±0.75 sec.	—
Injection Valve Differential Pressure	Differential	0-1000 psid	700 psid	-	±20 psi	100 milli sec.
Pump Minimum Flow Bypass	Flow Switch	0-15" H_2O	3" H_2O (4)	-	±120 gpm	250 milli sec.
Pump Dischage Pressure (signal to ADS)	Pressure Switch	10-240 psig	100 psig	-	±9 psi	—

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NOTES FOR TABLE 7.3-4 Page 2 of 2

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) 3" H_2O equal to 700 gpm.

(5) ~~Response time to be supplied later.~~ Deleted.

(6) ~~Information to be supplied later.~~
Margins will be ~~supplied~~ provided as part of the technical specifications.

(7) Required values will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.3-5

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM INSTRUMENT SPECIFICATIONS

PCRVICS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (8)	Required Accuracy (9)	Response Time (9)
Reactor Vessel Low Water Level	Level Switch	0-60" (1)	+12.5"	-	±3"	—
Reactor Vessel Low Water Level	Level Switch	-150/0/+60" (1)	-38"	-	±7.5"	—
Main Steam Line High Radiation	Radiation Monitor	See Table 7.2-1				
Main Steam Line Space High Temp	Thermocouple: Temperature Differential	50-350°F 0-150°F	(4)	-	±7°F ±3°F	2.25 sec.
Main Steam Line Low Pressure	Pressure Switch	50-1200 psig	840 psig	-	±15 psi at setpoint	1 sec.
Drywell High Pressure	Pressure Switch	0.2-6 psig	2 psig	-	±0.05 psi	.6 sec.
Containment Ventilization Exhaust High Radiation	Radiation Monitor	10^{-2} - 10^{2} mR/hr	(6)	-	±9.5 mR/hr	.5 sec.
Main Condenser Low Vacuum	Pressure Switch	0-30" hg ABS	23" hg ABS	-	±0.9" hg	—
Main Steam Line High Flow	Differential Pressure Switch	-15/0/150 psi	132.5 psi	-	±3 psi	.1 sec.

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NOTES FOR TABLE 7.3-5

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16 Technical Specifications for operational limits.

The range for safety related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16 Technical Specifications.

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) Trip settings will be established during preoperational testing (normally 50°F above ambient).

(5) Setpoint will be determined from instrument calibration curve.

(6) Setpoint will be established after background readings are determined during startup.

(7) ~~Response time to be supplied.~~ Deleted.

(8) ~~Information to be supplied later.~~
Margins will be supplied as part of the technical specifications.

(9) Required values will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

TABLE 7.4-1

REACTOR CORE ISOLATION COOLING INSTRUMENT SPECIFICATIONS

RCIC Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin	Required Accuracy (5)	Response Time (5)
Reactor Vessel High Water Level Turbine Trip	Level Switch	0-60" (1)	+55.5"	-	±3"	—
Turbine Exhaust High Pressure	Pressure Switch	0.5-80 psig	25 psig	-	±2 psi	—
RCIC System Pump Low Suction Pressure	Pressure Switch	30" Hg Vac. 10 psig	15" Hg Vac.	-	2 psi	—
Reactor Vessel Low Water Level	Level Switch	-150/0/+60" (1)	-38"	-	±7.5"	—
RCIC System Steam Supply Low Pressure (Reactor Pressure Low)	Pressure Switch	3-78 psig	50 psig	-	±30 psi	—
Turbine Overspeed	Centrifugal Device		125% of rated speed	-	±2% FS	—
Steam Supply High Differential Pressure	Differential Pressure Switch	0-400"H_2O 0-150"H_2O	225" H_2O 105" H_2O	- -	±8"H_2O ±3"H_2O	100 milli. sec. 100 milli. sec.

(1) Instrument zero equal to 527.5" above Vessel zero.
(2) See Chapter 16 Technical Specifications for operational limits.
(3) Trip settings shown are subject to change to comply with Technical Specifications.
(4) ~~Response time will be supplied later.~~ Deleted.
(5) Required values will be listed in the technical specifications. Listed values are subject to change to comply with the technical specifications.

Revised Draft Responses

for

31.006

31.014

31.026

31.047

Q. 031.006
(3.11.3)

The staff requests that the following information regarding the qualification test program be provided for Class 1E equipment: (a) the equipment design specificaiton requirements; (b) the test plan; (c) the test set up; (d) the test procedures; (3) the acceptability goals and requirements; and (f) the test results.

Provide this information for each of the following Class 1E components: (a) the 4.16 kV switchgear SM 7; (b) the damper operator for WMA-V-52C; (c) the fan WMA-FN-52B; (d) the logic equipment for the standby gas treatment system; (e) the diesel-generator control equipment; (f) the 480 V ESS switchgear MC-7A-A; and (g) the solenoid valve for the main steam line isolation valves.

RESPONSE:

An extensive seismic and environmental review program is presently underway encompassing BOP and NSSS scope, with a planned completion date in the second quarter of 1979.

Within the BOP scope, the equipment documentation has been extracted from the contract files, copied and categorized for easy retrieval. Within the NSSS scope, contract negotiations are underway with GE to perform a similar function.

A list of all Class 1E equipment including splices, terminal blocks, termination cabinets and connectors is presently being compiled. This list will contain the following information:

1. Equipment location
2. Safety functional requirement
3. Manufacturer & Model No.
4. Qualification Method (test-analysis)
5. Environmental Extremes
6. Identification and location of qualification documents

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

The completed list will be included in the FSAR as equipment tables in sections 3.10 (seismic) and 3.11 (environmental).

The extensive review program underway will also satisfy the requirements of IE circular 78-08.

Q. 031.014
(6.3.2.2)
(6.3.2.8)
(F 6.3-1a)

The location of sensors LS N001 A, B, C, and D, as shown in Figure 6.3-1a, does not appear to meet Seismic Category I requirements. Revise the design of the WNP-2 to assure that the senosrs controlling the transfer of suction to the suppression pool will be seismically and environmentally qualified for their location and environment.

RESPONSE:

Condensate storage tank pressure sensors used for level switches are designed and qualified to Seismic Category I requirements and are environmentally qualified.

The pressure sensors will be mounted on the interior side of the concrete fluid retaining walls surrounding the condensate storage tanks. The sensors will be located such that postulated failures of the condensate storage tanks will not compromise the sensors. The sensors are designed and located to withstand natural phenomenon, e.g., tornados and high winds, and will be freeze protected.

Q. 031.026
(7)

Describe the installation, operation, and removal of the "Startrek" computer system which is used for start-up testing of GE boiling water reactors, including the following topics: (a) specifications and qualification testing of electrical isolators; and (b) separation criteria for permanent and temporary wiring.

RESPONSE:

This system known as "Startrec" will be provided by General Electric on a loan basis for Startup Testing. This equipment will be located in the main control room and will consist of multiplexed data input terminals, data reduction; and data recording equipment.

The WNP-2 "Startrec" design implementation consists of both permanent and temporary equipment. Signal inputs will be permanently wired to test jacks located on the signal originating panel for non-safety related input sources and routed to a central divisionalized panel for signals originating in safety related equipment. This centralized panel will house isolation devices as well as output test jacks. The non-divisional wiring from all output test jacks will be temporary and routed overhead in the control room to the "Startrec" equipment.

All signals originating from safety related divisionalized equipment will be physically and electrically isolated such that faults occurring in the "Startrec" equipment cannot propogate back into safety related circuits. The isolation devices will be optical in nature, qualified to the standards of class IE equipment and meet the intent of Reg. Guide 1.75 concerning isolation devices. These isolation devices will be mounted in divisional centralized panels where all safety related equipment inputs will converge. There will be a central isolation panel for each division as required. The output of each isolated input point will be routed through test jacks to the "Startrec" equipment as non-divisional cabling.

In order to preserve plant availability all analog signal inputs to "Startrec" originating in non-safety related plant control system equipment will be electrically isolated through isolating amplifiers. This will prevent faults in the "Startrec" equipment from disturbing sensitive control system signals. The output of these amplifiers will be routed directly through test jacks to the "Startrec" equipment.

31.047
(7.5.2.4)

The seismic qualification of indicators and recorders for post-accident monitoring which is described in 7.5.2.4 is unacceptable. It is the staff's position that post-accident indicators and recorders must meet their minimum performance requirements before and after a seismic event without requiring adjustments or repair. (The staff acknowledges that these electro-mechanical devices may not provide accurate readings during severe vibrational excitation). Accordingly, we require you to provide a revised design which satisfies the staff's position on this matter.

RESPONSE:

The indicators and recorders are nonseismic. At the time WPPSS-2 was being designed, there was no IEEE standards or Regulatory Guide requirements for design of the subject instrumentation.

The instrumentation and readout devices are of high quality and from well-known manufacturers. This similar instrumentation is used for post-accident monitoring in such licensed and operating plants as Duane Arnold and Brunswick 2, and in such plants as Zimmer and LaSalle presently in the late stages of review. Therefore, it is the position of the General Electric Company that the instrumentaion provided is adequate.

Q. 031.064
(031.009)

Your response to Item 031.009 is incomplete. Describe the consequences of a turbine trip without bypass which is conincident with a postulated failure of the elctrical supply to actuate the "relief" function of the safety/relief valves. Describe all other loads which are powered from this particular electrical supply.

RESPONSE:

Is is considered highly unlikely that a turbine trip without bypass transient with coincident failure of the electrical supply to actuate the "relief" function of the safety/relief valves would occur. The peak pressure during a turbine trip without bypass transient with coincident failure of the "relief" function of the safety/relief valves is less severe than the safety relief valve sizing transient (i.e., MSIV closure with indirect scram) which only takes credit for the spring-action mode of operation of all safety/relief valves.) Furthermore, the peak fuel surface heat flux during a turbine trip without bypass occurs before the actuation of the safety/relief valves in their relief mode. Therefore, there is no effect on the MCPR limit during this transient if safety/relief valves fail to operate in their relief mode.

[illegible]

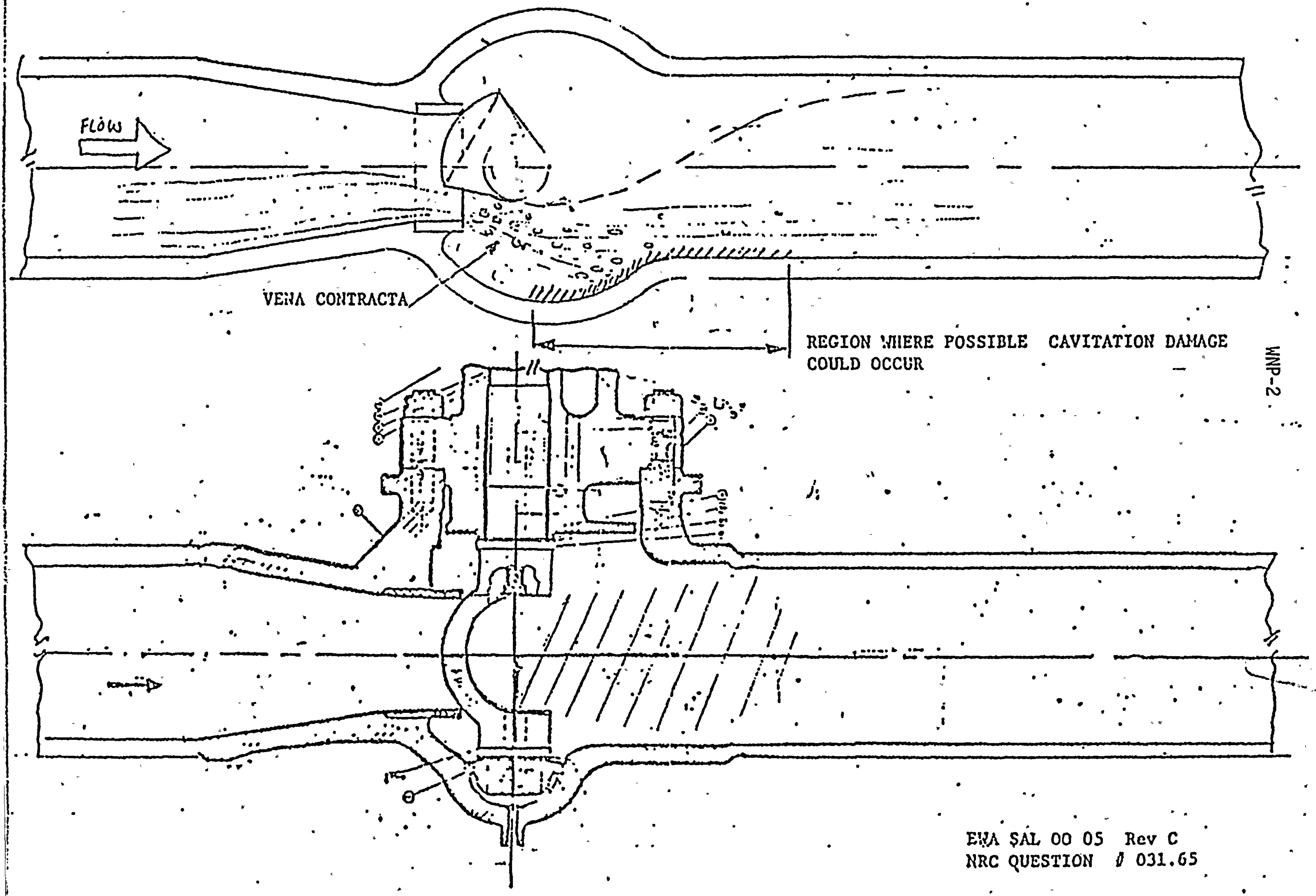
Q. 031.065
(031.010)

Your response to Item 031.010 is incomplete. Identify the specific recirculation flow control valve components which would be damaged by the collapse of bubbles in the pressure recovery areas of these valves.

RESPONSE:

In response to 031.065, enclosed is a figure which describes where possible cavitation damage could occur by the collapse of bubbles in the pressure recovery areas of the flow control valves.

Full cavitation, however, is a very unlikely event in the FCV. In addition, even if full cavitation occurs in the FCV, it would take 4 to 8 hours before the 0.003 in. corrosion allowance is worn away locally.



WNP-2

EWA SAL 00 05 Rev C
NRC QUESTION # 031.65

QUESTION 031.066

(031.009) (031.018) Your response to Item 031.018 appears to be based on design features which are not part of the BWR-5 design. Accordingly, provide the following additional information to support the validity of your response:

a. Describe any differences between the control rod scram speed and the resultant reactivity insertion rate for the WNP-2 facility and those of the GESSAR-238 design (Docket No. STN 40-447).

b. Clarify any discrepancies between the relief valve system design described in your response to Item 031.009 and the Class 1E relief valve design which is proposed in the GESSAR-238 design.

c. Describe the effects of the differences in the core size and physical characteristics of the WNP-2 reactor core and those of the Zimmer facility, on the validity of referencing the Zimmer study.

RESPONSE The response to Item 031.018 included the BWR-6 GESSAR study to give an overall view of the results for different product lines. The NRC has concluded that the offsite doses calculated in the BWR-6 GESSAR study are well within 10CFR100 limits. However, a more valid comparison would be one to the Zimmer plant, another BWR-5.

Parts (a) and (b) of Question 031.066 query the differences in reactivity insertion rate and relief valve system design. WNP-2 has the same control rod drive speed and resultant reactivity insertion rate as the Zimmer plant. The relief valve system design for WNP-2 is also similar to Zimmer, with the safety/relief valve capacities being equivalent.

By analyzing the limiting pressurization transient with concurrent failures of direct scram, the RPT and the bypass functions, it has been demonstrated in response to NRC question 221.359 on Zimmer Docket that the BWR 4/5/6 designs are each able to remain within the limits of 10CFR100. Even with widely varying design parameters, the results for each design were similar.

Thus, a like analysis solely for WNP-2 would provide results similar to that reached in the Zimmer and GESSAR study, i.e., that the offsite doses are well within 10CFR100 limits.

Q. 031.067
(RSP)
(031.025)

Your response to Item 031.025 indicates that the single failure of instrument bus 1A will result in a total loss of direct indication to the operator of a safe reactor shutdown. It is the staff's position that this is a unacceptable design. Accordingly, we require you to provide a design which will satisfy the single failure criterion.

RESPONSE:

The system has been changed to provide essential power to the full core display and the scram valve position indicators. A power panel which is fed from a diesel generator upon loss of offsite power is now the source rather than instrument BUS 1A. In addition, the operator has alternate diverse means of determing a reactor trip as follows:

1. Rod position display from the rod sequence control system.
2. Process computer trip and positon log.
3. R.P.S. system annunciation.
4. Local mechanical pressure indicators for the scram valve pilot air header.
5. Neutron monitoring flux indicators.

The redesign and diverse means of verifying reactor shut down meet the single failure criteria.

Q 031.068

Provide the wiring diagrams for the instrument racks for the WNP-2 facility which are comparable to the diagrams cited in the Item 031.032. It is the staff's position that these must be made available so that they may be audited to determine whether the required electrical separation of independent divisions has been achieved.

RESPONSE:

The following drawings are provided as requested in 031.068, (7 copies).*

Rack	Number	Sheet	Revision
IR-10	E538	7	3
IR-10	757-E-673		B
IR-11	E538	8	1
IR-11	757-E-674		E
H22-P004	E539	21	1
H22-P004	127D1827TC		2
H22-P005	E539	2	2
H22-P005	127D1837TC		2
H22-P026	E539	4	2
H22-P026	828E390TC		2
H22-P027	E539	3	1
H22-P027	828E387TC		2

* Packaged and sent separately.

Telecopied to W?9/55 1/29/78

Q 031.069
(3.8.2.1)
(6.2.1.1)
(031.001)

Your response to Item 031.001(p) is incomplete. Describe the air supply, pressure control, and position indication for the butterfly valves in accordance with the guidance provided in Section 7.3.1 of Regulatory Guide 1.70. Clarify the reference to 6.2.1.1.2 in the response to Item 031.001(p) since this response does not address the staff's concern regarding the position indication instrumentation.

Response

Please refer to 3.8.2.1.3, 6.2.1.1.2c and 7.3.1.1.2.9.1 for the information requested.*

*See the attached draft markup.

TABLE OF CONTENTS (Continued)

LIST OF FIGURES (Continued)

3

DR Appendix E

WNP-2 Amendment No. 2

12/21/78 1:55

Figure Number	Title	Engineering Dwg. No.
3.8-56	Class I Air Supply System for Containment Vacuum Breaker Valves	M 619, Sh. 161
3.8-57	~~HVAC~~ Containment ~~Purge~~ Vacuum ~~Supply Isolation~~ Breaker Butterfly Valve Logic Diagram	M 620 Sh. 543-4
3.8-58	Containment Drywell/Wetwell Vacuum Breakers - Control Logic Diagram	M 620 Sh. 543-11

Appendix E

7.3.1.1.2.6 System Sequencing

A discussion of all sequencing of all subsystem of the PCRVICS is provided in 7.3.1.1.2.4.

7.3.1.1.2.7 System Bypasses and Interlocks

A manual bypass of the main steam line low-pressure signal is effected in the startup mode of operation (see 7.3.1.1.2.4.1.5.6).

7.3.1.1.2.8 System Redundancy and Diversity

The variables which initiate isolation are listed in the circuit description, 7.3.1.1.2.4.1. Also listed there are the number of initiating sensors and channels for the isolation valves.

7.3.1.1.2.9 System Actuated Devices

To prevent the reactor vessel water level from falling below the top of the active fuel as a result of a pipeline break, the valve closing mechanisms are designed to meet the minimum closing rates specified in Table ~~7.3-13~~ 6.2-16.

The main steam line isolation valves are spring-closing, pneumatic, piston-operated valves. They close on loss of pneumatic pressure to the valve operator. This is fail-safe design. The control arrangement is shown in Figure 7.3-4. Closure time for the valves is adjustable between 3 and 10 seconds. Each valve is piloted by two three-way, packless, direct-acting, solenoid-operated pilot valves both powered by ac. An accumulator located close to each isolation valve provides pneumatic pressure for valve closing in the event of failure of the normal air supply system.

The sensor trip channel and trip logic relays for the instrumentation used in the systems described are high reliability relays. The relays are selected so that the continuous load will not exceed 50% of the continuous duty rating. Table 7.3-7 lists the minimum numbers of trip channels needed to ensure that the isolation control system retains its functional capabilities.

7.3.1.1.2.9.1 Primary Containment ~~Purge Supply~~ Vacuum Breaker ~~Valves~~ System.

The operation, controls, and position indication for the wetwell-to-drywell and the reactor building-to-wetwell vacuum breaker system is described in 3.8.2.1.3.

3.8.2.1.3 Description of Vacuum Relief System

Refer to Figure 9.4-8 for an illustration of the vacuum relief system described below.

Three 24-inch reactor building-to-wetwell vacuum relief lines, each containing a 24- inch vacuum breaker valve and an automatic air operated butterfly valve, are provided between the reactor building and the suppression chamber. These valves prevent ~~a~~ excessive vacuum from developing in the primary containment vessel from such causes as inadvertent containment spray actuation.

Each butterfly valve is air-operated to close. During normal plant operation each valve is maintained in a closed position by means of control air supply through a 3-way solenoid pilot valve. The plant control air supply system is backed up by a Quality Class I air supply system (see Figure 3.8-56). Upon venting of air from the air-operators, the butterfly valves are spring-actuated to open. This is accomplished through remote manual deenergization of the solenoid pilot valve, or by a signal from the differential pressure switch which deenergizes the solenoid pilot valve when the secondary containment atmospheric pressure is more than 0.5 psi greater than the suppression chamber atmospheric pressure.

Two limit switches, wired to indicator lights in the control room, are provided with each butterfly valve for position indication. One switch actuates when the valve is fully open, and provides an alarm and "open" visual indication in the control room. The other switch actuates when the valve is fully closed, and provides a "closed" visual indication in the control room. See Figure 3.8-57 for an illustration of the logic, controls, and position indication for the butterfly valves.

In series with each butterfly valve is a single disk check valve. The disk is maintained in the closed position during normal operation by means of a spring actuated lever arm and magnets embedded in the periphery of the disk. The magnetic and spring forces are overcome, and the disk opens when the pressure differential across the valve ~~exceeds 0.2 psi~~ within the range of 0.15 to 0.35 psi. The disk is fully open when the pressure difference is 0.5 psi. In addition, air cylinders are provided for remote operation of the disk. Compressed air is supplied by the plant control air system~~.~~ which is backed up by a Quality Class I air supply system. Each disk has two air cylinders, one to open and one to close the disk. Each air cylinder is actuated through energization of a 3-way solenoid pilot valve. The two solenoid pilot valves associated with each disk are operated by a remote manual switch in the control room. During normal operation the remote manual switches are in the neutral position and the solenoids are deenergized. Each valve disk is provided with ~~three~~ Five ~~limit switches~~ sensors for position indication. Contact probes for ~~two~~ four of the ~~limit switches~~ are mounted ~~180°~~ 90° apart on the disk face, and are wired to indicator lights in the control room to provide open and closed position indication. The ~~third limit switch~~ fifth contact probe, also wired to a light in the control room, indicates when the disk is fully open. See Figure 3.8-58 for an illustration of the controls and position indication for the vacuum breaker check valves.

Nine 24-inch wetwell-to-drywell vacuum relief valves attached to the downcomers in the suppression chamber are provided to return non-condensibles from the wetwell to the drywell to prevent too large an upward pressure differential across the diaphragm floor after a loss-of-coolant accident.

Each wetwell-to-drywell vacuum relief valve assembly consists of two disks and seats which operate independently. The operation, controls, and position indication for each disk is as described above for the single disk check valves, except that the control air system is not backed up by a quality Class I air supply system. since the air system is only required for testing. The vacuum breaker valves are sized to ensure that the following design conditions are not exceeded:

a. The drywell internal design pressure of 2.0 psi below reactor building pressure.

b. The suppression chamber internal design pressure of 2.0 psi below reactor building pressure.

c. The upward design pressure difference across the diaphragm floor of 6.4 psi.

The design evaluation for the vacuum relief is discussed in 6.2.1.1.4.

The vacuum relief valves are designed to ASME Section III, Subsection NC for Class 2 components.

Electrical systems associated with the control and position indication for the vacuum breaker valves ~~is~~ are Class 1E.

is made in Table 1.3-4. The water stored in the suppression pool is capable of condensing the steam displaced into the wetwell through the downcomer vents, and the amount of water is sufficient so as not to require any operator action for at least ten minutes immediately following initiation of a LOCA. In addition, the design allows the water from any pipe break within the primary containment to drain back to the suppression pool. This "closed loop" ensures a continuous, adequate supply of water for core cooling.

c. Negative Loading

The primary containment is designed for the following negative loadings:

1. A drywell pressure of 2.0 psi below reactor building pressure

2. A wetwell pressure of 2.0 psi below reactor building pressure

3. An upward pressure across the diaphragm floor of 6.4 psid.

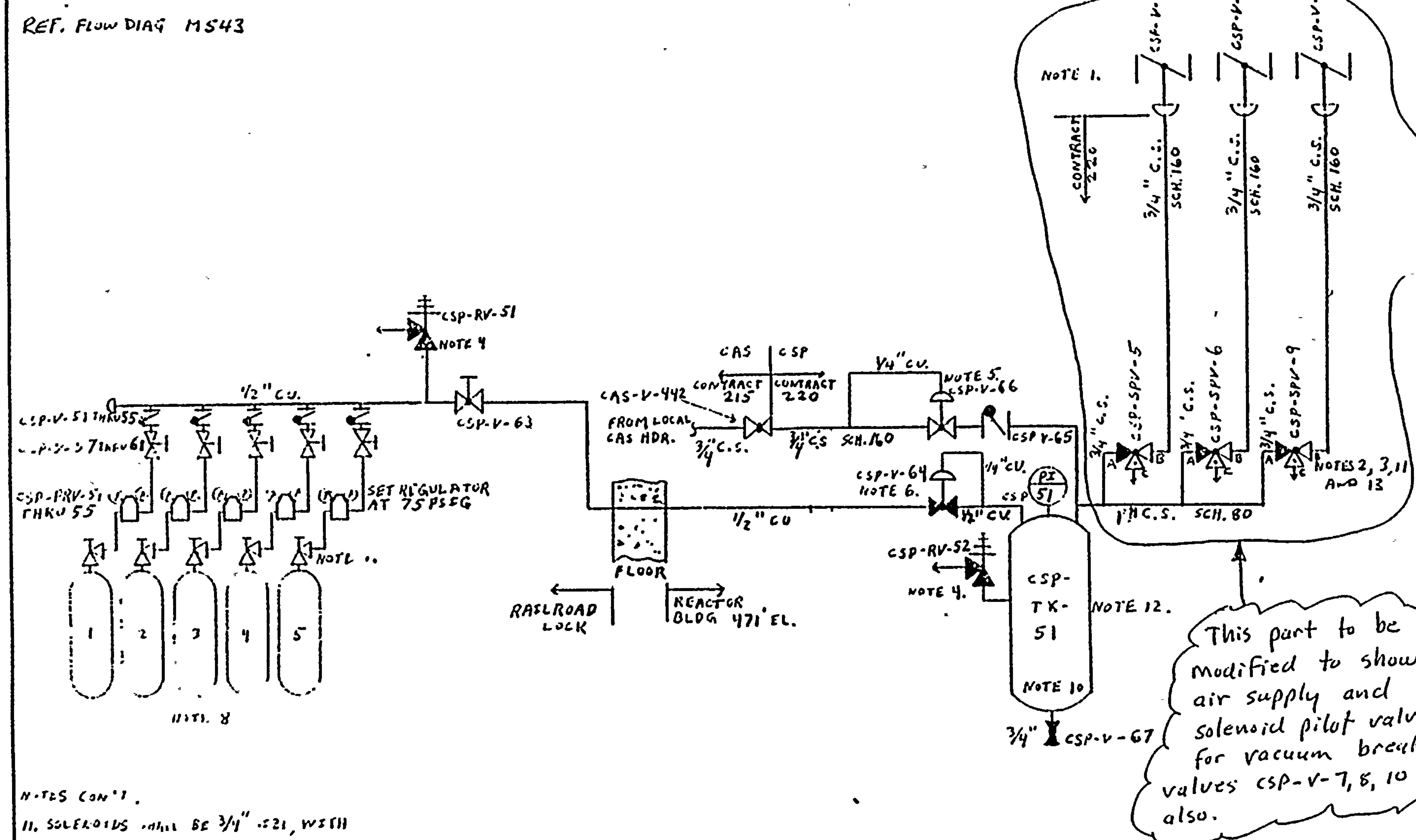
The nine 24" wetwell-to-drywell (WW-DW) and the three 24" reactor building-to-wetwell (RB-WW) vacuum breaker lines are sized to ensure that the above negative loadings are not exceeded. The vacuum breaker systems are described in 3.8.2.1.3. Instrumentation reference can be found in 7.3.1.1.2.9.1.

The primary containment is designed for a total external pressure of 4 psid; however, since the compressed insulation between the concrete biological shield and the containment exerts a uniform 2 psid external pressure - half of the total external pressure differential allowed - the drywell pressure may be no less than 2 psi below the reactor building pressure.

d. Environmental Conditions

The means to maintain the required environmental conditions inside the primary containment during normal operation is discussed in 6.2.1.1.8. With the exception of energy removal from the suppression pool, there are no requirements for

REF. FLOW DIAG M543



NOTES.

1. VALVES CSP-V-5, CSP-V-6, AND CSP-V-9 ARE NORMALLY CLOSED, FAIL OPEN ON LOSS OF AIR.
2. SOLENOIDS ARE SHOWN IN DEENERGIZED POSITION. SOLENOIDS ARE NORMALLY ENERGIZED.
3. ROUTE DISCHARGE TO WITHIN 1' OF FLOOR.
4. SET RELIEF VALVE AT 150 PSIG.
5. SET VALVE CSP-V-66 TO CLOSE WHEN SENSING LINE PRESSURE FALLS TO 75 PSIG.
6. SET VALVE CSP-V-64 TO OPEN WHEN SENSING LINE PRESSURE FALLS TO 75 PSIG.
7. SUPPLIED WITH ASSOCIATED EQUIPMENT.
8. AIR BOTTLES (1.54 CU.FT) SUPPLIED BY OTHERS. CONTRACTOR IS TO SUPPLY BOTTLE RACK AND CONNECTION PIGTAILS.
9. ALL EQUIPMENT AND PIPING ON THIS SHEET, EXCEPT CAS SYSTEM SHALL BE QUALITY CLASS I, SEISMIC CATEGORY I ASME III, CLASS 3

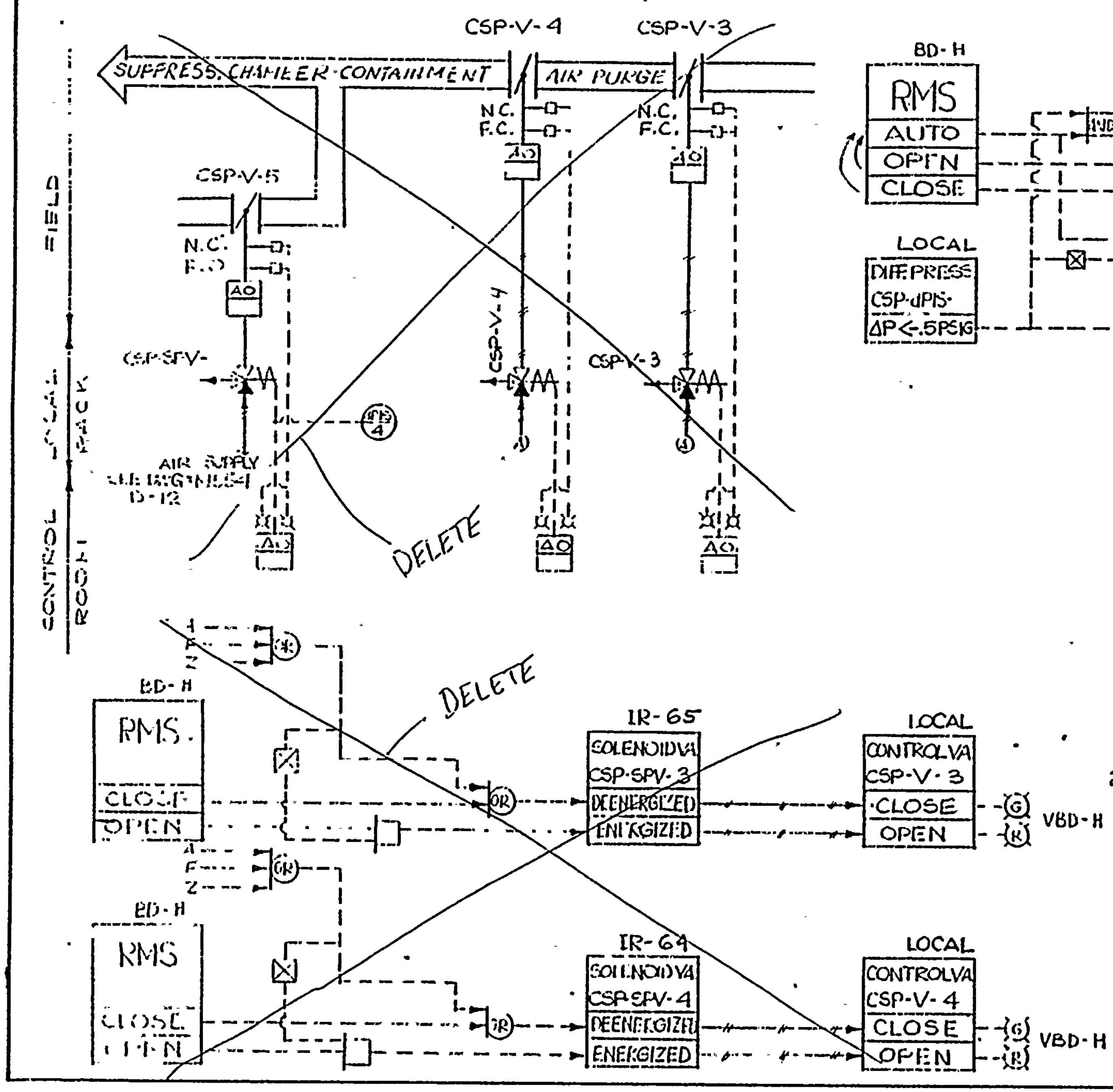
NOTES CON'T.

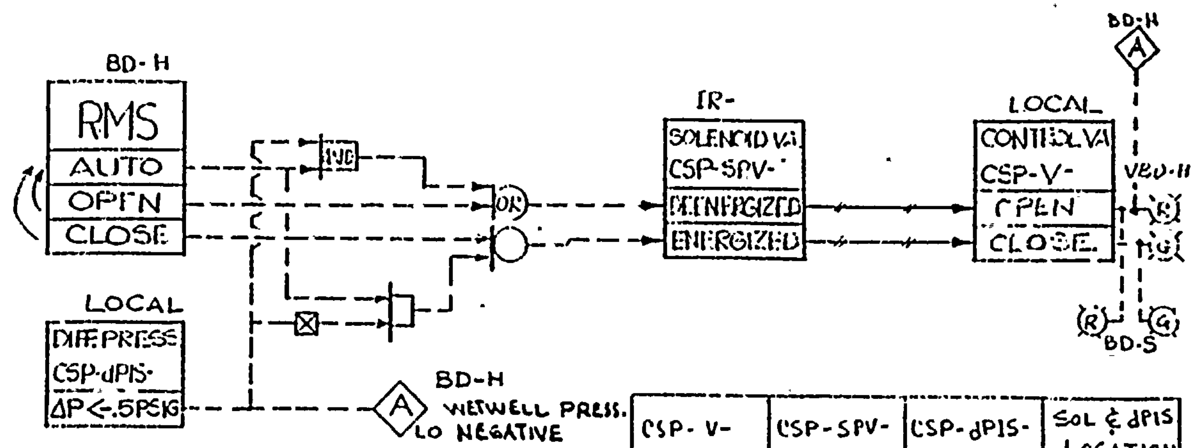
11. SOLENOIDS SHALL BE 3/4" SIZE, WITH MIN. NO. OF 2. SOLENOIDS SHALL BE BUBBLE-TIGHT IN BOTH POSITIONS AND TESTED TO ENSURE THIS.
12. SECURE TANK TO BOTH FLOOR AND WALL
13. SOLENOID VALVES CSP-SPV-5, 9 ARE MOUNTED ON IR-64 AND CSP-SPV-6 IS MOUNTED ON IR-63.

CLASS I AIR SYSTEM FOR CONTAINMENT VACUUM BREAKER VALVES.

FIG. 3.8-56

~~FIG. 7.3-17~~





CSP-V-	CSP-SPV-	CSP-dPIS-	SOL & dPIS LOCATION
-5	-5	* -5	IR-64
-6	-6	** -4	IR-63
-9	-9	* -6	IR-64

* USE CONTACTS IN SERIES FOR SINGLE ANNUNCIATOR DROP ON BOARD H2

** ANNUNCIATOR DROP ON BOARD H1.

~~NOTES:~~

~~1 ISOLATION SIGNAL CODES.~~
~~A-REACTOR VESSEL LOW WATER LEVEL (DIV. I & II)~~
~~F-HIGH DRYWELL PRESSURE (DIV. I & II)~~
~~Z-HIGH RADIATION REACTOR BUILDING VENTILATION SYSTEM (DIV. I & II).~~

~~2. FOR REFERENCES SEE DWG. # M 543~~

FIG. 3.8-57
~~FIG. 7.3.22~~

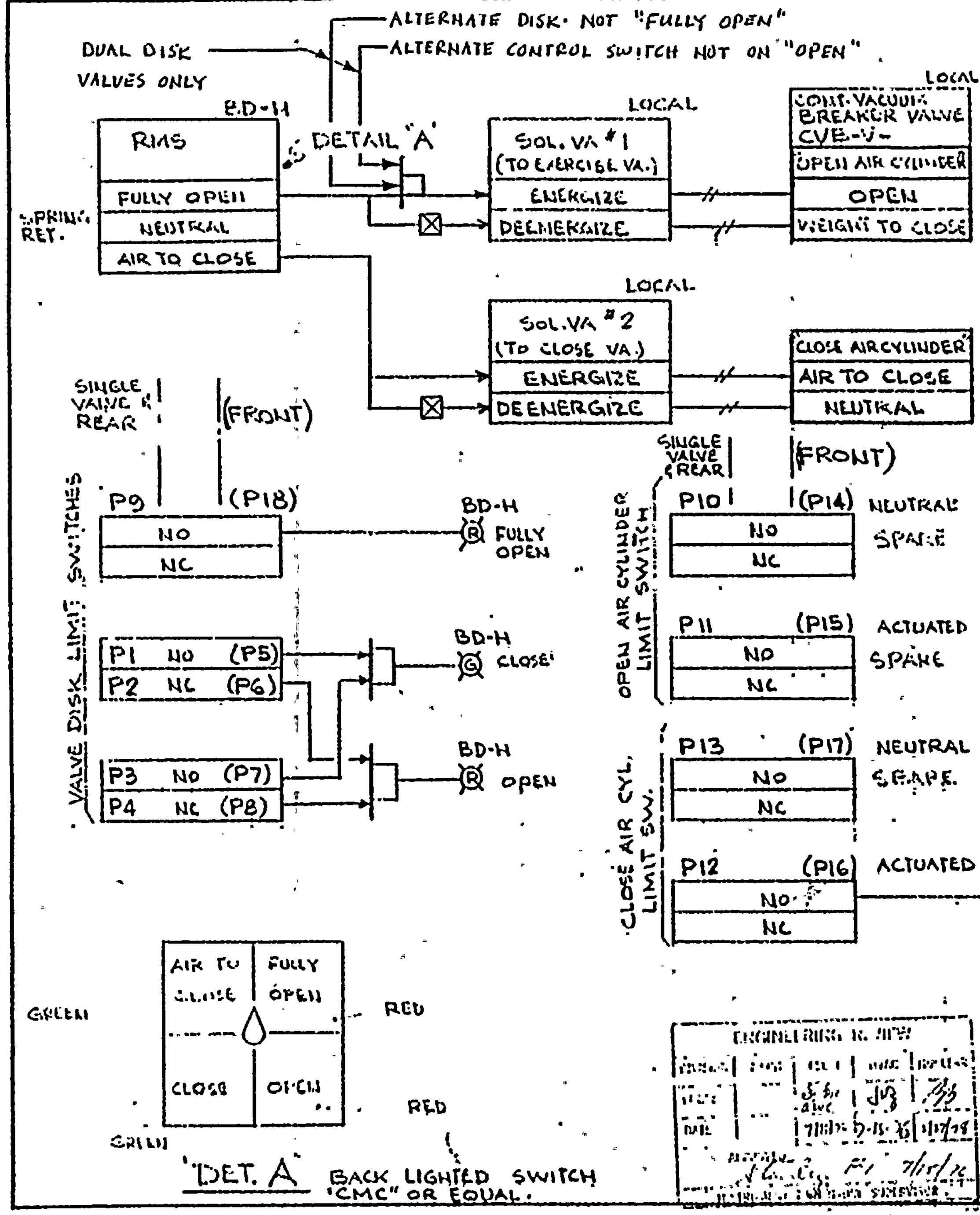
~~HVAC~~ CONTAINMENT VACUUM BREAKER ~~PURGE SUPPLY~~ BUTTERFLY ~~ISOLATION~~ VALVE LOGIC DIAGRAM

PROJECT "WPPSS"
HANFORD Nº 2
W.O. NO. 2808-24
BY J. JUEL 12/3/74

BURNS AND ROE, INC.

I & C DWG. NO. M-

SHEET 543-11 OF



RMS	SOL. VA #1	LOCATION	SOL. VA #2	LOCATION	VACUUM BREAKER VA	REMARKS
CVB-V-1A	CVB-SPV-1A1	LOCAL	CVB-SPV-1A2	LOCAL	CVB-V-1AB	
1B	1B1		1B2			
1C	1C1		1C2		1CD	
1D	1D1		1D2			
1E	1E1		1E2		1EF	
1F	1F1		1F2			
1G	1G1		1G2		1GH	
1H	1H1		1H2			
1J	1J1		1J2		1JK	
1K	1K1		1K2			
1L	1L1		1L2		1LM	
1M	1M1		1M2			
1N	1N1		1N2		1NR	
1P	1P1		1P2			
1Q	1Q1		1Q2		1QR	
1R	1R1		1R2			
CSP-V-7	CSP-SPV-7A		CSP-SPV-7B		CSP-V-7	
CSP-V-8	CSP-SPV-8A		CSP-SPV-8B		CSP-V-8	
CSP-V-10	CSP-SPV-10A		CSP-SPV-10B		CSP-V-10	
CVB-V-1S	CVB-SPV-1S1		CVB-SPV-1S2		CVB-V-1ST	
CVB-V-1T	CVB-SPV-1T1		CVB-SPV-1T2			

NOTE: FOR VALVES, AIR OPERATOR CYLINDERS, SOLENOID VALVES AND LIMIT SWITCHES SEE "ANDERSON GREENWOOD DWG. Nº 04-3852, 3802 (B&R FILE Nº 213-00-1720, 1750.)

FIG. 3.8-58

1 2/20/78 PCN-4427

ENGINEERING REVIEW

CONTAINMENT (DRYWELL/WETWELL) VACUUM BREAKERS CONTROL LOGIC DIAGRAM

Q. 022.018 (6.2.1)

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

Q. 031.070 (RSP) (6.2.2.2) (6.5.2.2) (7.3.1.1) (031.001) (031.011)

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

Response:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

References

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

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

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

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

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

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

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

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

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

6.2.1.1.5.4 Analytical Results

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

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

[INSERT]

6.2.1.1.6 Suppression Pool Dynamic Loads

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

6.2.1.1.7 Asymmetric Loading Conditions

See comment in 6.2.1.1.6.

(Insert for page 6.2-30)

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

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

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

6.2.7 REFERENCES

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

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

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

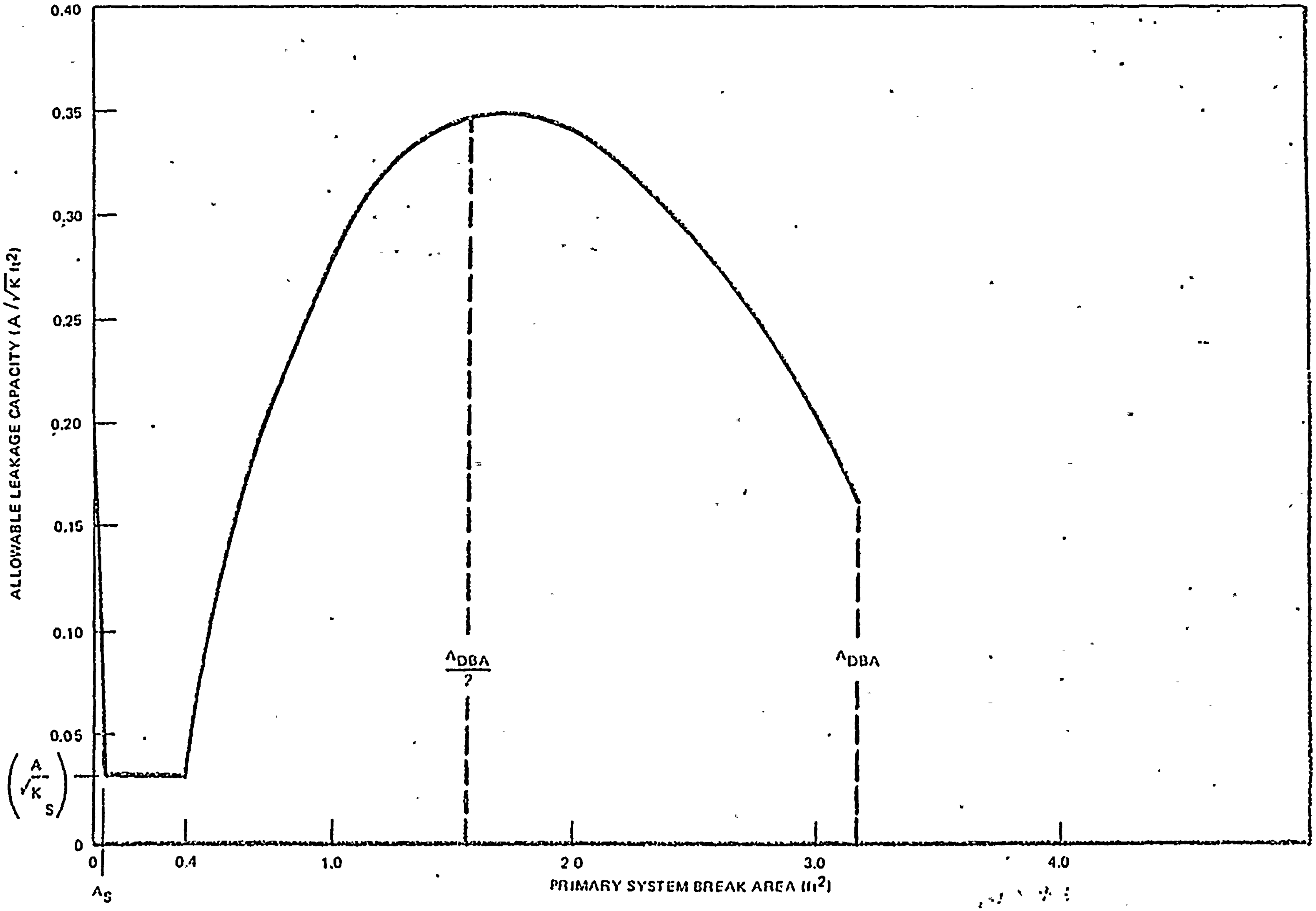
6.2-4 "MK II Containment Dynamic Forcing Functions Information Report (Revision 2)", General Electric and Sargeant and Lundy, NEDO-21061, September 1976.

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

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

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

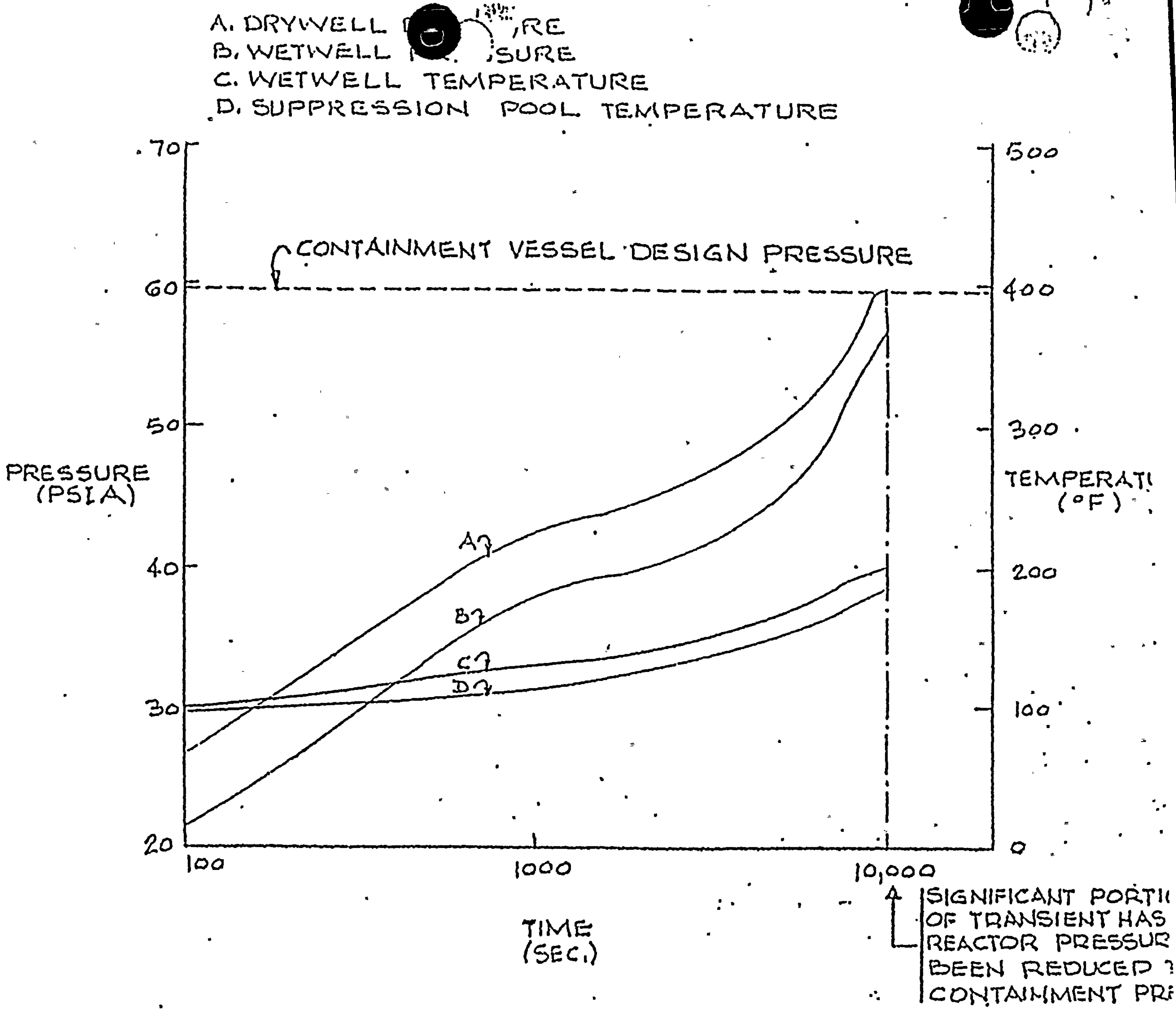
6.2-8 Wheat, L. L.; Wagner, R. J.; Niederauer, G. F.; Obenchain, C. F., CONTEMPT-LT--A COMPUTER PROGRAM FOR PREDICTING CONTAINMENT PRESSURE-TEMPERATURE RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, ANCR-1219, Aeroject Nuclear Company, June, 1975.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

ALLOWABLE LEAKAGE CAPACITY

FIGURE 6.2-17



N PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

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

FIGURE 6.2-17b

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTAINMENT TRANSIENT FOR A/√K = 0.0045 FT²

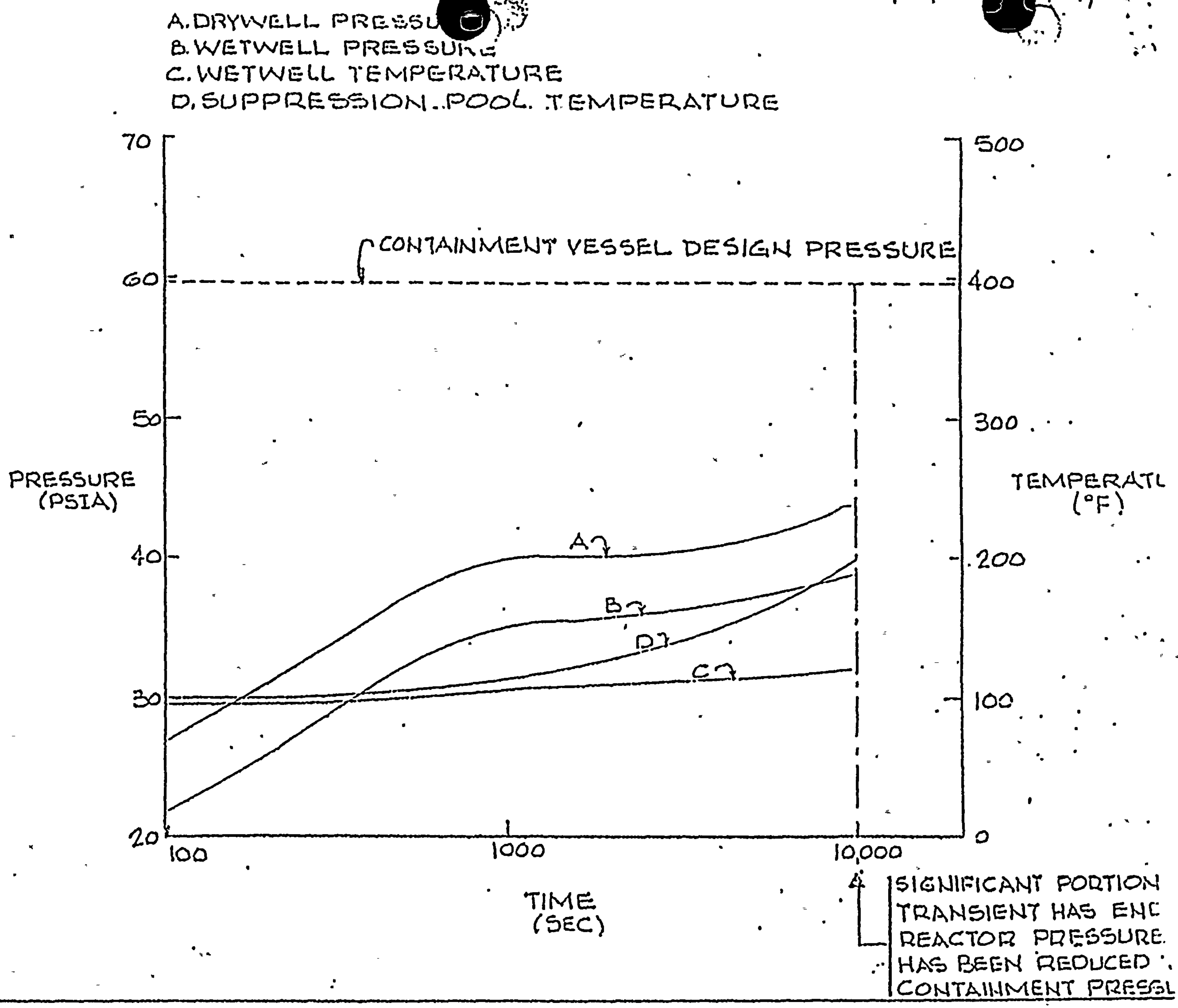


FIGURE 6.2-17c

Q. 031.071
(7.3.1.1)
(031.001)

The primary containment and reactor vessel isolation control system receives power from the reactor protection system motor generator sets. Describe how the reactor operator determines the position of each motor operated and each solenoid operated or controlled isolation valve after a loss of the motor generator sets. We are concerned that your present design de-engergizes these sets during a loss of offsite power and does not include automatic restart of the motors.

RESPONSE:

Status indication for all motor operated containment isolation valves is powered from diesel generator busses and thus is not dependent on the availability of the RPS M/G set busses.

In addition to valve position status located on the main bench board at the control switch for each isolation valve, a complete containment isolation valve position display exists on Board S. This panel is located in the first row of control room back panels. The power sources used for the display valve position status are uninterruptible with diesel generator backup. Thus, the solenoid operated isolation valves also have position indication that is not dependent on RPS M/G set availability.

Q. 031.072
(7.6.1.5)
(F 7.7.1)
(F 7.2.5)
(031.007)
(031.045)

Your response to Item 031.001(t) is unacceptable. Clarify the discrepancies between 7.1-11, Figures 7.1-1 and 7.2-5, and Section 7.6.1.5. Specifically, clarify the number of instruments and the designation of these instruments with regard to their trip channel assignments. Provide justification for running redundant signals through the same penetrations.

RESPONSE:

There are no discrepancies between Table 7.1-11, Fig. 7.1-1 and 7.2-5, and Section 7.6.1.5. Also, redundant signals do not run through the same penetrations.

The neutron monitoring system trip outputs to the reactor protection system are derived from 6 APRM channels as follows:

RPS trip channel	APRM
A1	A&E
A2	C&E
B1	B&F
B2	D&F

This combination which uses APRM channels E&F in redundant RPS trip logic allows an APRM channel in each trip division to be bypassed by the operator without the loss of ability to Scram on a high flux condition. See FSAR Section 7.2.1.1.4.2 for additional information concerning the neutron monitoring system inputs to RPS.

Q. 031.073
(9.2.5.3)
(F9.3-13)
(031.007)
(031.045)

Clarify the discrepancy between the responses to Items 031.007 and 031.045. In this regard, the staff notes that:

a. Section 9.3.5.3 states, "The pumps and valves are powered and controlled from separate buses and circuits so that a single failure will not prevent system operation."

b. The separate buses are powered from onsite sources.

c. Figure 9.3-13 shows both heaters to be powered from a single source, and

d. The heaters in the standby liquid control system of previously reviewed similar facilities are powered from two separate Class 1E buses.

Response 031.073:

No discrepancy exists between the responses to 031.007 and 031.045; section 9.3.5.3 reflects the WNP-2 design. However, clarification is as follows:

General Electric has shown pumps and valves powered and controlled from separate power sources for the purpose of optimizing availability wherever possible. There is no requirement that the SBLC system, within its own system boundary, meet single failure criteria. Single failure criteria is satisfied on an intersystem basis with the SBLC system being a back-up to the CRD system.

The heaters may be powered from two separate buses but this is not the requirement. The heaters are not, nor intended to be functionally redundant. One is a mixing heater and the other is an operating heater. After initial mixing is completed, the mixing heater is deenergized. The temperature of the sodium pentaborate solution can then be reduced to a lower than mixing temperature without crystallization. The operating heater serves to maintain sufficient holding temperature, but is seldom initiated because the ambient temperature of the tank environs is most often above the heater's trip point.

Q. 031.074
(RSP)
(7.7.1.2)
(F7.7-3)
(031.001)

The response to Item 031.001 (bb) is incomplete. While additional information on the reactor manual control system is provided in Section 7.7.1.2 and Figure 7.7-3, this figure does not present a design which is technically feasible (e.g., Figure 7.7-3b, Zone L/11). Accordingly, we require you to replace Figure 7.7-3 with a set of drawings which presents a technically feasible implementation of this system for the WNP-2 facility.

Response:

The purpose of Figure 7.7-3 is to describe the reactor manual control system logic. It is not intended as a basis for assessing technical feasibility. The elementary level drawings identified in Table 1.7 should be used for any detail review of the design.

QUESTION 031.075

(7.7.1.3) (7.7-7) (031.008) The response to Item 031.008 and the additional information which is presented in Figure 7.7-7 and 7.7.1.3 is incomplete. In this regard, the staff notes that the design includes an interlock which prevents the transfer of one pump from high speed operation to low speed operation when the second pump is operating at high speed and the control switch is moved to the motor generator position. Explain why this interlock is provided. Justify not providing a similar interlock in the pump start circuitry in order to prevent a similar occurrence under the same conditions (i.e., both pumps running at high speed) if the switch should be placed in the start position.

RESPONSE

(031.075) The purpose of the recirc pump interlock identified in 031.075 is to prohibit flow inbalance between the pumps, thereby minimizing jet pump vibration. A similar interlock in the pump start circuitry is not provided because, by design, if both pumps are running at high speed and the switch is placed in the start position, nothing will happen to affect the status of either pump.

Q. 031.076
(RSP)
(6.7)
(7.3.2.3)
(Q.031.019)

It is the staff's position that neither the information which is provided in response to Item 031.019 nor the information which is presented in 6.7 and 7.3.2.3 provides sufficient information on the main steamline isolation valve leakage control system. Describe this system in accordance with the guidance provided in Section 7.3 of Regulatory Guide I.70, including a process and instrumentation drawing, an electrical schematic, and a failure mode and effects analysis which is sufficiently detailed to address failures at the component level. For example, describe the consequences of a spurious closing of the contacts of relay K4 under all plant-operating modes, including testing.

RESPONSE:

In addition to FSAR sections 6.7 and 7.3.2.3 and the MSIV leakage control system instrumentation and controls is described in section 7.3.1.1.3. The system is shown diagramatically in P&ID form in FSAR figure 3.2-25 with logic diagrams shown in FSAR figures 7.3-18a-g.

Electrical schematics for the MSIV leakage control system (drawings, E519, shts 30 & 31) have previously been submitted as part of FSAR section 1.7.

A complete FMEA will be completed in Feb. 1979.

The MSIV leakage control system does not contain a relay K4.

Q. 031.077
(031.021)
(031.037)

The responses to Items 031.021 and 031.027 are unacceptable because the accuracy of the instrumentation sensors is not provided. The information provided in Table 7.2-1 is labeled as that which is required in contrast to that which is actually provided. Additionally, the first item of Table 7.2-1 indicates that a pressure switch which has an error of plus or minus 10 psi is required. In similar BWR-5 application, this instrument is stated to have accuracy of plus or minus one percent of full scale. Accordingly:

a. Provide an amended response to Item 031.021 which includes the accuracy of the sensors which are installed in your plant.

b. Provide an amended response to Item 031.037 which defines such terms as "adequate margin" and describes the criteria and procedures for determining and adjusting the instrument test frequencies.

RESPONSE:

See the response to question 31.063 (31.016).

Q. 031.78
(031.050)
(T7.3-5)

Clarify the discrepancy between the response to Item 031.050(a) and Table 7.3-5. Specifically, explain how the temperature trips, which are usually set between 135 to 185 degree F, can reliably detect a 50 gpm leak from the reactor heat removal system during a prolonged cold shutdown when the primary coolant temperature may be less than 135 degrees F.

RESPONSE:

The equipment area temperature monitors are not intended to detect 50 gpm leakage from the emergency core cooling systems (ECCS) during the long-term recovery following the postulated loss-of-coolant accident. There is insufficient energy in the ECCS fluid (when the reactor is depressurized and cooled down) to heat the equipment area in the event of the postulated 50 gpm leak.

The design bases for the ECCS equipment area temperature monitors is to identify leakage and provide inputs for isolation in the event of leakage from high energy RCPB lines beyond the second isolation valve during normal plant conditions.

In addition to the termperature sensors, RHR flow, Reactor water level, HPCS and LPCS flow detectors listed in response to Q. 31.050, numerous drain flow indicators are provided. These consist of:

a. Restricting orifice, placed directly in the collecting drain header. These orifices are designed and calibrated to pass 5 gpm with a static head of 6 inches. As soon as total flow in the collector exceeds this flow rate an electrode will sense the increase of fluid and activate an annunicator alarm.

b. Conductance type electrodes 6 inches long mounted in a suitable fitting threaded into the collector-header.

c. Mating control switch consisting of a solid state electronic relay to operate controls.

These instruments are shown on Flow Diagrams (Figures 9.3-5, 9.3-6 and 9.3-8). The arrangement provides a fail safe operation, because the restrictive orifices are supplied for each floor elevation, independently, in the common collecting header. This forms a series arrangement. Should any elevation instrument fail, flow will collect at the next level and provide annunciation in the Main Control Room.

Q. 031.79
(7.3)
(031.039)

Provide the following additional information with regard to the control of the minimum flow bypass for each emergency core cooling system pump:

a. Describe how the flow sensor is calibrated and tested.

b. Describe how proper operation of the relay it controls is tested.

c. Describe how proper functioning of the bypass valve is verified during the periodic test since these valves will cycle in less than 1 minute under the test conditions.

d. Describe how the design of the bypass valve control subsystems satisfy the requirements of: (1) GDC-21; (2) IEEE Std 279 - 1971, Sections 4.10, 4.12, and 4.20; and (3) IEEE Std 338 - 1971.

Additionally, describe how the design of these subsystems conform with the guidance provided in Regulatory Guides 1.22 and 1.118.

RESPONSE:

All ECCS Minimum Flow Return Line Control Valves operate in the same manner, i.e., to protect pumps from overheating these valves open when system flow is below a specified value and close when flow exceeds this value.

a. The flow sensors (differential pressure switches) for all ECCS Minimum Flow Return Line Valve Controls are calibrated and tested during reactor operation in the same manner as other safety related instrumentation. That is, they are valved out of service and a calibrated test pressure introduced. This allows the flow set point to be adjusted per design requirements and also verifies mechanical switch operability.

b. and c.

Proper operation of the relay(s) controlled by the flow sensor and proper functioning of the minimum flow valve itself are demonstrated during the periodic full flow system test. The operator initiates the system with the test return line valve closed and verifies that the minimum flow valve is open. The test return line valve is opened slowly and by reading the system flow indications on the main bench board the flow at which the minimum flow return valve begins to close is verified. This point is identified through the use of the valve position indicating lamps as well as the alarm provided by the process computer which monitors the control signal to close the valve.

When coming back down in flow proper operation again can be shown by verifying that the minimum flow valve reopens at the correct flow point. This is again accomplished through the use of the system flow indication, valve position indication and the process computer alarm.

d. As described in Parts a, b, and c the intent of GDC-21, IEEE STD. 279-1971 Para. 4.10 & 4.20, IEEE STD. 338-1971, Regulatory Guides 1.22, and 1.118 is met with the present design. Reference to Para. 4.12 of IEEE STD. 279-1971 is apparently in error. There is no connection between the Minimum Flow Return Line Valve Operation and the standard's requirements regarding "operating bypasses".

Reactor Systems

Branch

Responses

Q. 212.1

Your discussion of internally generated missiles does not include the potential for damage to safety systems and/or the generation of secondary missiles inside containment as a result of a falling object. The discussion also does not address the potential for the failure of safety systems inside containment caused by secondary missiles generated by postulated primary missiles impinging on a component or structure inside containment. Provide a discussion of these two matters.

RESPONSE:

Please refer to 3.5.1.2.4 and 3.5.1.2.5 for the information requested.*

*See attached draft.

3.5.1.2.4 Falling Objects

Structural elements, equipment, and components inside containment which could be considered as potential falling objects are supported to satisfy Seismic Category I requirements. The only exceptions to this are the monorail hoists inside containment which are not analyzed for Seismic I loading conditions, but are chained in place while not in use to ensure that they do not become falling objects which could damage safety systems. On this basis, falling objects are not postulated in the methodology described in 3.5.1.1.2. The physical separation and redundancy of safe shutdown systems as described in 3.5.1.2.2 also assures that falling objects do not present a threat to these systems.

3.5.1.2.5 Secondary Missiles Generated by Postulated Primary Missiles

The design objectives for providing protection against postulated primary missiles inside primary containment are implemented through physical separation and redundancy of safety systems, as discussed in 3.5.1.2.2.

The potential for the failure of safety systems inside primary containment caused by secondary missiles generated by postulated primary missiles impinging on a component or structure inside primary containment depends on the mass, velocity, trajectory, and other physical characteristics of the postulated secondary missile. These in turn depend on similar parameters which define the postulated primary missile, as well as the assumed failure mechanism which results in secondary missile formation. The conclusions to be drawn from any such analysis would be questionable, in view of the many assumptions involved. For this reason, the methodology used in evaluating the effects of primary missiles, as described in 3.5.1.1.2, is not appropriate for secondary missiles. The plant design features which provide protection against primary missiles and assurance that primary missile formation will not occur serve also to eliminate or at least minimize the potential for failure of safety systems caused by secondary missiles. These design features include the following:

a. Physical separation of redundant systems.

b. Reorientation of postulated primary missiles.

c. Prevention of primary missile ejection.

d. Provision of missile barriers.

E. Fredenburg
12-4-78

Q. 212.2

Provide a discussion of leakage between systems with respect to back leakage to the Emergency Core Cooling Systems (ECCS) through isolation valves and show conformance with the guidance contained in Regulatory Guide 1.45.

RESPONSE:

The isolation valves between the RCPB and ECCS lines are high quality, redundant, Class 1 valves, with position indication in the main control room. As stated in Regulatory Guide 1.45, "substantial intersystem leakage from the RCPB to other systems across passive barriers or valves is not expected." If intersystem leakage between the RCPB and ECCS should occur, it will be contained within the system. Since the ECCS systems are kept full of water, leakage will increase system pressure so that ECCS pressure indicators and high pressure control room alarms will alert the operator to potential overpressurization of the system from excessive leakage.

QUESTION 212.003 (6.3)

Your discussion of single failure does not adequately address ECCS passive failures during long-term cooling. Accordingly, provide a response to the attached Reactor Systems Branch Technical Position regarding the leak detection requirements for passive failures in the ECCS piping.

REACTOR SYSTEMS BRANCH TECHNICAL POSITION

Leak Detection Requirements for ECCS Passive Failures

The passive failures to be considered are limited to leaks from valve stem packing and pump seals. The sum of these leak rates may range from essentially no leakage up to the equivalent of the sudden failure of the seal of the largest ECCS pump (e.g., about 50 gpm). It is the staff's position that detection and alarms be provided to alert the operator of passive ECCS failures during long-term cooling. The timing of these alarms should be such that the reactor operator has sufficient time to identify and isolate the faulted ECCS line. Provide the following information regarding the ECCS leak detection system:

a. An identification and justification of the maximum leak rate;

b. The maximum allowable time for operator action, including a justification of the time interval;

c. A demonstration that the leak detection system will be sensitive enough to provide an alarm to the operator, subsequent identification by the operator of the faulted line, and, finally, permit the operator to isolate the faulted line prior to the leak creating any undesirable consequences such as flooding of redundant equipment. The minimum time to be considered for this sequence of events is 30 minutes.

d. A demonstration that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

Additionally, the ECCS leak detection system must meet the following standards: (1) control room alarm; and (2) IEEE-279, except single failure requirements.

RESPONSE:

The response to this question will be submitted in Feb. 1979.

Q. 212.4 (6.3)

Provide a list of all valves in the ECCS and their positions during normal operation, all ECCS modes of operation, and all shutdown modes of operation. The valve identification should be consistent with that shown on the process and instrumentation control diagrams.

RESPONSE:

This information is contained in the valve position tables on the ECCS process diagrams, Figure 6.3-2 (HPCS), Figure 6.3-6 (LPCS) and Figure 5.4-14b, c (LPCI/RHR).(a draft of Fig. 5.4-14b is attached).

RESIDUAL HEAT REMOVAL SYS

731E966AD

FCF238X184AD (E12-1020)

MODE F

(SEE NOTE 3)

POSITION ○	1	2	3	4	5	6	43	7	8	9	10	13	24	1
FLOW-GPM	-	7450	—	—	—	—	—	—	—	—	—	—	7450	-
PRESS-PSIA	147	X	X	X	X	X	X	X	X	X	X			147
TEMP °F		[illegible]	—	—	—	—	—	—	—	—	—	—	[illegible]	-
MAX. PRESS DROP-FEET	[illegible]	—	—	—	[illegible]	—	—	—	—	—	—	—	—	—

MODE A-1

(SEE NOTE [illegible])

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW GPM	—	7450	—	—	—	—	—	—	—	—	—	7450	—
PRESS PSIA	[illegible]	X	X	X	X	—	—	—	—	—	—		[illegible]
TEMP °F	—	[illegible]	—	—	—	—	—	—	—	—	—	[illegible]	
MAX PRESS DROP FEET	—	—	—	—	[illegible]	—	—	—	[illegible]	—	—	[illegible]	—

MODE G

R. PRESS 135 PSIG

MODE A-2

(SEE NOTE 10)

MODE S

MODE B

MODE S (CONT'D)

MODE C-1

R. PRESS 1000 PSIG

MODE C-3

MODE C-2

R. PRESS 1000 PSIG

MODE C-4

MODE C-2 (CONT'D)

MODE D

(SEE NOTE 13)

MODE E

THIS DRAWING SUPERSEDES DRAWING 731E966C REV.3 FOR HANFORD 2 (DIFFERENCES ARE DENOTED BY ENCIRCLEMENTS).

731E966AD

FIG 5.4-14b

POSITION ○	31	32	4	5	6	7	8	9	10	47	16	46	N	7	71	18	19	20	8	32	4	5	6	9	10	46	N	23	[illegible]	26	[illegible]	[illegible]	16	52	27
DESIGN PRESS. IN PSIG	125	220		[illegible]	500							1250		500		[illegible]	500			220		[illegible]	500			1250		[illegible]	220				500	[illegible]	
DESIGN TEMP IN °F	212	356			350						100	575		350			350			355			212			575			350				460		
ESTIMATED LINE SIZE	24"				8"							12"		8"			16"			24"			8"			12"		20"		18"	24"		12"		
	LPCI LINE (VIA RHR HX BYPASS)													HX LINE						LPCI LINE (DIRECT TO REACTOR)								SHUTDOWN SUCTION					SHUTDOWN RETURN LINE		

SEE NOTE 10

POSITION ○	34	35	36	37	17	18	13	20	38	39	40	401	41	42	14	30	31	54	66	40	402	44	47	46	21	16	52	23	15	53	44	45	24
DESIGN PRESS. IN PSIG	1250				500		500				500		125		500			[illegible]		600	125		500			500			500	125			
DESIGN TEMP IN °F	575				480		480				140		140		350					140	212		356	212		356	212		356	212			
ESTIMATED LINE SIZE	10"	8			16		16"	4"							6"					4"			16"			4"			18"				
	STEAM TO HX						CONDENSATE FROM HX								HEAD SPRAY LINE					CONDENSATE TO POOL			DRYWELL SPRAY			SUPPRESSION POOL SPRAY			SYSTEM TEST LINES (LOOPS A & B)				

POSITION ○	56	561	4	45	451	43	6	61	62	63	6	61	63
DESIGN PRESS IN PSIG	220			500	125		500	125			500	125	
DESIGN TEMP IN °F	355			212			212				356	212	
ESTIMATED LINE SIZE	18"			18"			4"			4"	4"		
	SYSTEM TEST LINES (LOOP C)						LINE FROM BYPASS LINES (LOOP C)				LINE FROM BYPASS LINE		

TABLE I - VALVE POSITION CHART

SEE NOTE 6

O - VALVE OPEN
T - VALVE THROTTLED
P - STRAINER ...
BLANK SPACE INDICATES VALVE IS CLOSED

TABLE [illegible] LINE LOSSES

NOTE A	6-14 PUMP DISCHARGE LINE TO RPV FLOODING PENETRATION
NOTE B	NONE
NOTE B	1 2-3 (SUCTION LINE SUPPRESSION POOL TO PUMP), 47 21 & 16 23 (CONTAINMENT SPRAY LINES PUMP DISCHARGE TO SPRAY HEADERS)
NOTE C1	43-44 (HX CONDENSATE TO SUPPRESSION POOL), 41-42 (HX CONDENSATE LINE TO RCIC LINE)
NOTE C2	20 40 41 HX CONDENSATE TO RCIC LINE POINT 41
C ZONE	NONE
NOTE C3	33 34 35 STEAM LINE FROM RCIC LINE TO HX HEADER
NOTE C4	35 17 (STEAM LINE HEADER TO HX)
NOTE D	14 54 31 29 (VESSEL HEADER SPRAY LINE) 19 25 5 (SHUTDOWN SUCTION LINE RPV TO PUMP) 56 4 (SHUTDOWN SUCTION LINE TEST BRANCH TO PUMP)
NOTE E	16 27 (SHUTDOWN DISCHARGE LINE LPCI BRANCH TO RECIRC LINE)
NOTE F	13 53 24 44 45 45 (TEST LINE TO SUPPRESSION POOL)
NOTE G	6 63 (PUMP MINIMUM BYPASS LINE)

RESIDUAL HEAT REMOVAL SYSTEM

FOR REVISIONS, SEE SH 1

FIG 5.4-14

Hydrology - Meteorology

Branch

Responses*

*includes update of previous question

Q. 371.6

Provide the results of a transient analysis to determine the adequacy of the ultimate heat sink spray ponds under emergency conditions, including consideration of the requirements for both the temperature and volume of the water. (Refer to Regulatory Guide 1.27, Rev. 2, for guidance on this matter.) Provide the basis for any assumptions used in your analysis and a discussion of your analytical techniques.

Response:

The mass loss and thermal transient analyses for the UHS following a design basis LOCA are presented in 9.2.5. The analysis is presently under revision and will be complete by January 31, 1979. The revised information will be submitted in an amendment during the first quarter of 1979.

Q. 372.7

Provide the maximum and minimum values of the wet and dry bulb temperatures used for the design of safety-related equipment. Provide the basis for these maximum and minimum temperatures.

RESPONSE:

It is understood that the above question is directed towards the ultimate heat sink. As discussed in 2.3.1.2.3 and 9.2.5 the WNP-2 UHS was analyzed using the following maximum temperatures that occurred on July 10, 1975:

1. Maximum dry bulb temperature: 105.71^{o}F (FSAR Table 2.3-7d)

2. Maximum wet bulb temperature: 73.96^{o}F (FSAR Table 2.3-7d)

This maximum wet/dry bulb temperature combination, as discussed in 2.3.1.2.3, results in the worst pond thermal performance combined with the other temperatures occurring that day.

The ultimate heat sink freeze protection design is adequate to protect the systems to the lowest recorded temperatures at the site. This minimum dry bulb temperature is -27^{o}F (FSAR Table 2.3-1). Minimum wet bulb temperature was not used in the design of the ultimate heat sink.