REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8212170173 DOC.DATE: 82/12/09 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
AUTH.NAME AUTHOR AFFILIATION
BOUCHEY,G.D. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Responds to NRC Question 010.066 re pipe break in BWR scram sys (NUREG-0803).Response to Questions 010.067 - 010.074 will be submitted under separate cover during Dec 1982.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 60 SIZE: 13
TITLE: Licensing Submittal: PSAR/FSAR Amdts & Related Correspondence

NOTES:

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	NRR/DL/ADL		1	0	NRR LB2 BC		1	0
	NRR LB2 LA		1	0	AULUCK,R.	01	1	1
INTERNAL:	ELD/HDS2		1	0	IE FILE		1	1
	IE/DEP EPDS	35	1	1	IE/DEP/EPLB	36	3	3
	NRR/DE/AEAB		1	0	NRR/DE/CEB	11	1	1
	NRR/DE/EQB	13	2	2	NRR/DE/GB	28	2	2
	NRR/DE/HGEB	30	1	1	NRR/DE/MEB	18	1	1
	NRR/DE/MTEB	17	1	1	NRR/DE/QAB	21	1	1
	NRR/DE/SAB	24	1	1	NRR/DE/SEB	25	1	1
	NRR/DHFS/HFEB	40	1	1	NRR/DHFS/LQB	32	1	1
	NRR/DHFS/OLB	34	1	1	NRR/DL/SSPB		1	0
	NRR/DSI/AEB	26	1	1	NRR/DSI/CPB	10	1	1
	NRR/DSI/CSB	09	1	1	NRR/DSI/ICSB	16	1	1
	NRR/DSI/METB	12	1	1	NRR/DSI/PSB	19	1	1
	NRR/DSI/RAB	22	1	1	NRR/DSI/RSB	23	1	1
	REG FILE	04	1	1	RGN5		3	3
	RM/DDAMI/MIB		1	0				
EXTERNAL:	ACRS	41	6	6	BNL(AMDTS ONLY)		1	1
	DMB/DSS (AMDTS)		1	1	FEMA-REP DIV	39	1	1
	LPDR	03	1	1	NRC PDR	02	1	1
	NSIC	05	1	1	NTIS		1	1

TOTAL NUMBER OF COPIES REQUIRED: LTTR 52 ENCL 45

[illegible]

[illegible] 1986.

[illegible]

[illegible]	[illegible]	[illegible]	[illegible]

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

December 9, 1982
G02-82-975
SS-L-02-CDT-82-110

Docket No. 50-397

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

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
RESPONSE TO NRC QUESTION
010.066, NUREG-0803

Reference: Letter, A. Schwencer (NRC) to R.L. Ferguson (SS), "WNP-2 FSAR - Request for Additional Information", dated August 10, 1982

Enclosed are sixty (60) copies of the response to NRC Question 010.066, which was transmitted to us via the reference letter. This response addresses open item number 20, "Pipe Break in BWR Scram System".

The responses to the remaining Questions 010.067 - 010.074 will be submitted under separate cover during December 1982.

Very truly yours,

GD Bouchey

G. D. Bouchey
Manager, Nuclear Safety and Regulatory Programs

CDT/jca
Enclosure

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

B001
1/60

8212170173 821209
PDR ADOCK 05000397
A PDR

10.66 (4.6) NUREG-0803 states that pipe breaks in the control rod drive hydraulic system and the resulting environmental effects should be verified on a plant specific basis. In order to conform to the guidelines of NUREG-0803, provide information to address the following concerns:

1. Taking no credit for seals and assuming no operator actions inside of the control room for 20 minutes (30 minutes for the first operator action outside of the control room plus five minutes for each additional action), provide the following information for a non isolable break in the CRD piping between the containment penetration and the first isolation valve.

Reactor Coolant	- Mass flow rate out of the break as a function of time (= f (t))
	- Temperature = f (t)
Compartment	- Temperature = f (t)
	- Pressure = f (t)
	- Humidity = f (t)
	- Airborne Radioactivity level = f (t)

Provide the assumptions used in determining the above information. If a computer was used, provide the computer printout and the following information:

1. With respect to the pipe to be broken:

   a. Type of fluid (water or steam);

   b. Temperature;

   c. Pressure;

   d. Source of the fluid;

   e. Flow rate (or assumed flow rate);

   f. Pipe internal diameter;

   g. Wetted perimeter of the break (feet);

   h. Total pipe internal volume;

   i. Exit flow area, if the break was not in the pipe, just described above;

   j. Area of flow restriction, if any;

   k. Differential elevation from the source to the pipe break;

l. Total flow resistance (only if the fluid is water);

m. Means to stop fluid flow (none, gate valve, globe valve, etc.); and

n. If l.m is a valve, then the valve's open throat area, full open flow coefficient, valve closure time, and delay time until initiation of valve closure.

2. With respect to the compartments being analyzed;

a. Number of compartment analyzed; and

b. For each compartment:

i. initial temperature
ii. initial pressure
iii. initial humidity
iv. free air volume (cubic feet.
v. number of vents and vent areas (square feet) for each vent; and
vi. minimum pressure to initiate flow to the next compartment (psia).

3. All assumptions used, including but not limited to the:

a. Orifice coefficient for the "end effects" for the discharge fluid; and

b. Fluid expansion factor,

II. Verify that all electrical and mechanical equipment needed to mitigate the event is qualified to the environmental conditions determined in Part I. Verify that no pump cavitation will occur when pumping 212°F water.

III. Provide a discussion of the procedural steps to be taken to isolate the break in the CRD pipe at the outside surface penetration of containment to terminate the small pipe LOCA accident. Identify all equipment and materials required to isolate the break. Provide a commitment to maintain on site these items as dedicated equipment and materials. Discuss your procedure to verify periodically the existence and condition of these dedicated items.

IV: Assuming that the sumps are inoperative for the event in Part I (since they are not seismic Category I, Class 1E), provide maximum water level in the compartment. Verify that no equipment required to bring the plant to a safe condition will fail as the result of internal flooding. Verify that no personnel radiation hazard will exist due to wading through reactor coolant.

V: Verify that all analysis performed in Parts I and IV includes time for the items listed below. The first action outside of the control room should be assumed to be at least 30 minutes after annunciation in the control room plus five minutes for each personnel action in accordance with ANS 58.8. Installing scaffolding requires multiple actions just like donning protective garb.

1. HP survey of the area and documentation;

2. Establishment of protective garb requirements;

3. Establishment of change areas, clean areas, check-in and check-out lists, waste disposal facilities and transportation of necessary garb to the area for the workers;

4. Following all HP procedures;

5. Review of repair procedures.

Assume that the event occurs with the minimum plant personnel available on any shift.

RESPONSE TO NRC QUESTION 10.66(4.6)

INTRODUCTION

Following the Supply System submittal responding to NUREG 0803 concerns (Reference 1 and 2), the NRC is requesting, via NRC Question 10.66(4.6), that the Supply System evaluate a single CRD withdrawal line break at a specified location. However, according to the NRC's own NUREG 0803:

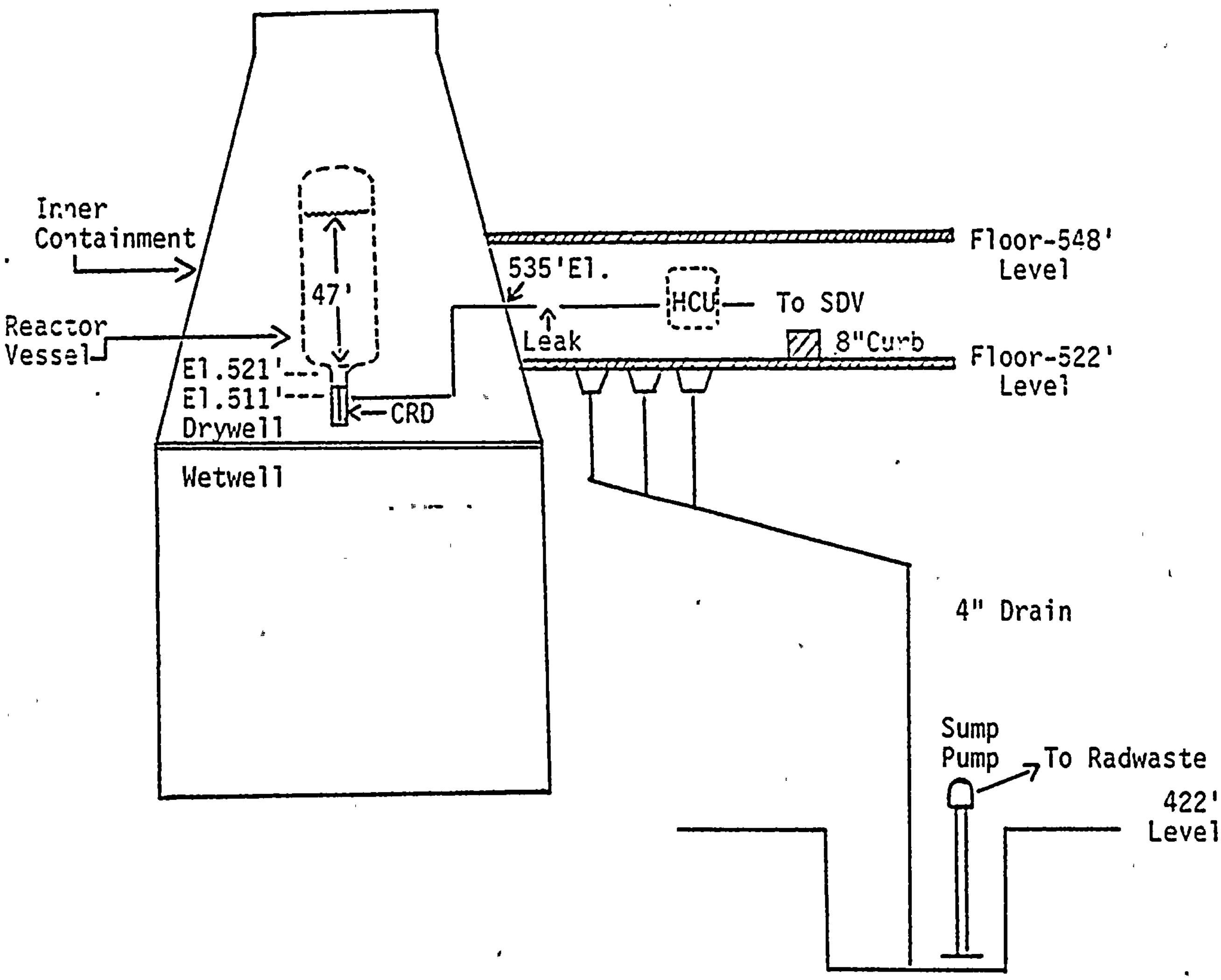
> From page 2-2 . . . This assessment is based on the fact that the CRD withdraw lines penetrating the containment and routed to the HCUs are small in diameter (3/4 in.) and are conservatively designed and of high quality. Nevertheless, even if the staff postulated a break in one of these lines during reactor operation (including scram):
>
> (1) The leakage through this break is within the reactor coolant makeup capabilities (feedwater and reactor core isolation cooling) since, as required by GDC 14, "Reactor Coolant Pressure Boundary," the CRD system contains redundant CRD seals and a restricted flow area that limits the reactor coolant leakage to a very small value;
>
> (2) The reactor can be shut down and cooled down in an orderly manner; and
>
> (3) No leakage from the SDV, where flow from all other CRD withdraw lines is accumulated following scram, will occur through the break because of the existence of a check valve between the SDV and the withdraw line manual isolation valves.
>
> and from page 4-27. . . Breaks upstream from the isolation valves in the 3/4 in. piping were judged to be minor in size and with no potential short-term effects on the core cooling capability.

The Supply System agrees that a single CRD withdrawal line rupture event is minor and fully enveloped by existing analyses. For example, an Instrument Line Rupture, FSAR Sections 15.6.2 and 6.3.3 presents a similar type break, but with larger leak flow and more severe consequences. However, as requested in NRC Question 10.66 (4.6), evaluation of the single CRD Withdrawal Line Rupture is presented in the following paragraphs to provide the NRC reviewers with quantification of the event to support the already recognized conclusions stated above.

Question 10.66 (4.6) contains five parts (I through V) and each section or part is addressed separately below:

FIGURE 1

Sketch of WNP-2 Elevations



NRC QUESTION 10.66, PART I

I. Taking no credit for seals and assuming no operator actions inside of the control room for 20 minutes (30 minutes for the first operator action outside of the control room plus five minutes for each additional action), provide the following information for a non isolable break in the CRD piping between the containment penetration and the first isolation valve.

Reactor Coolant	- Mass flow rate out of the break as a function of time (= f (t))
	- Temperature = f (t)
Compartment	- Temperature = f (t)
	- Pressure = f (t)
	- Humidity = f (t)
	- Airborne Radioactivity level = f (t)

Provide the assumptions used in determining the above information. If a computer was used, provide the computer printout and the following information:

1. With respect to the pipe to be broken:

   a. Type of fluid (water or steam);

   b. Temperature;

   c. Pressure;

   d. Source of the fluid;

   e. Flow rate (or assumed flow rate);

   f. Pipe internal diameter;

   g. Wetted perimeter of the break (feet);

   h. Total pipe internal volume;

   i. Exit flow area, if the break was not in the pipe, just described above;

   j. Area of flow restriction, if any;

   k. Differential elevation from the source to the pipe break;

   l. Total flow resistance (only if the fluid is water);

m. Means to stop fluid flow (none, gate valve, globe valve, etc.); and

n. If 1.m is a valve, then the valve's open throat area, full open flow coefficient, valve closure time, and delay time until initiation of valve closure.

2. With respect to the compartments being analyzed:

a. Number of compartment analyzed; and

b. For each compartment:

i. initial temperature
ii. initial pressure
iii. initial humidity
iv. free air volume (cubic feet)
v. number of vents and vent areas (square feet) for each vent; and
vi. minimum pressure to initiate flow to the next compartment (psia).

3. All assumptions used, including but not limited to the:

a. Orifice coefficient for the "end effects" for the discharge fluid; and

b. Fluid expansion factor.

RESPONSE TO PART I:

A simplified diagram of the postulated line rupture is shown in Figure 1. It is assumed that a complete severence of a withdrawal line occurs, following a reactor scram, at a location between the Reactor Containment and the first isolation valve (in HCU). There is no back-flow through the break due to a check valve between the break location and the scram discharge volume (SDV). Therefore, the only leak out of the break is the reactor coolant that can leak around the CRD piston seals. This leakage is normally about 2 gpm and can be as high as 5 gpm with badly worn seals. The stipulation in the Question 10.66 above that no credit for seals be taken is an unrealistic conservatism. However, GE tests show that with seals completely missing, at operating pressure, the leakage is a maximum of 10 gpm. This value is used as the initial leak rate.

The scenario can be defined as follows:

Time	Event Description
0	Reactor scrams. Complete severence of a single CRD withdrawal line occurs.
20 min.	Operator initiates cooldown at 100oF/hr due to leakage exceeding tech spec values.
240 mins (3.8 hr)	Reactor on RHR with RC temp$\leq$ 200oF.

The analysis assumptions and results are presented below in the same order and numbering system used in the question.

1. a. Type of fluid: Water

   b. Temperature: } See Figure 2

   c. Pressure: }

   d. Source of fluid: Reactor vessel

   e. Flow Rate: See Figure 3

   f. Pipe internal diameter: }

   g. Wetted Perimeter: }

   h. Total pipe internal volume: } Not applicable, See Item 1.j below.

   i. Exit flow area: }

   j. Area of flow restriction: 1.19 x 10-4 ft2 - equivalent area for 10 gpm leak at 1000 psia.

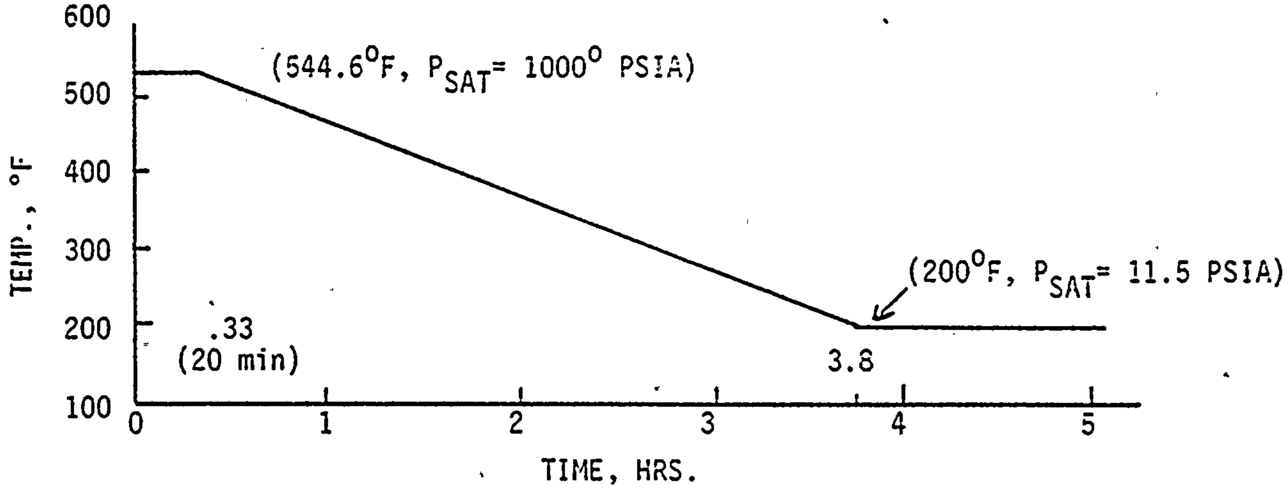


FIGURE 2: Temperature Versus Time With 100° F/HR Cooldown

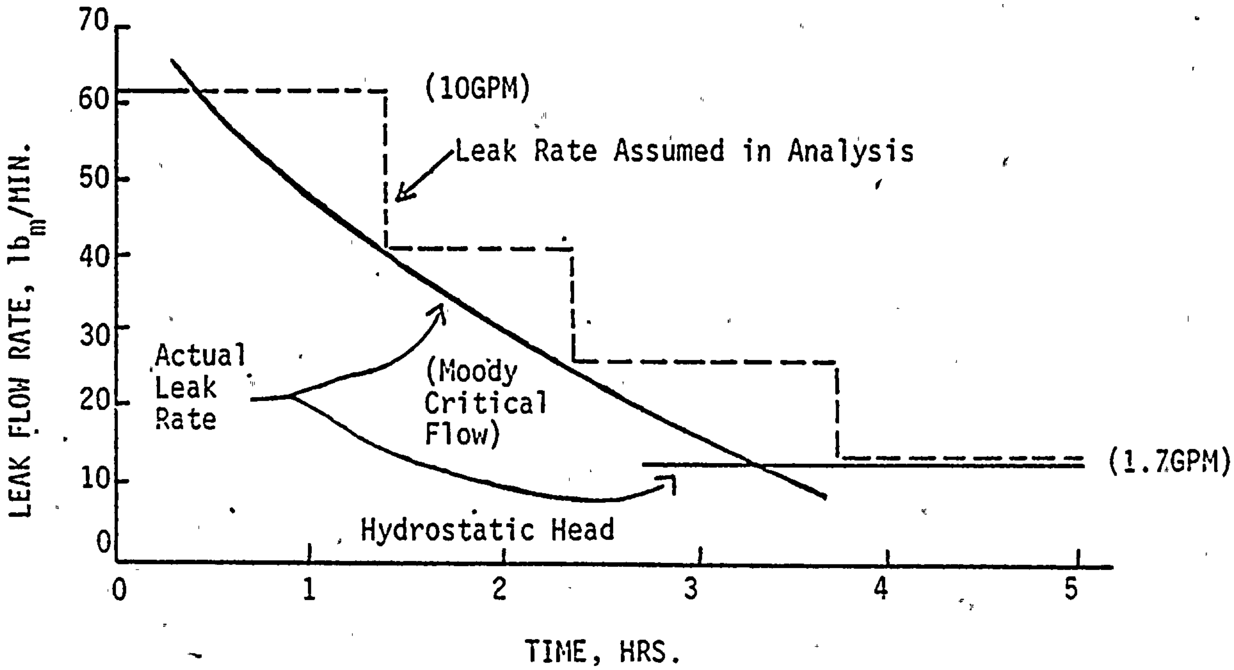


FIGURE 3: Leak Rate Versus Time

k. Differential elevation: See Figure 1.

l. Total Flow Resistance: Total pipe resistance was neglected for the low flow rates encountered. The hydrostatic analysis accounted for orifice pressure drop at the CRD, see Item 3.a below.

m. Means to stop fluid flow: Mechanical plugging at operator's discretion, see Part III below.

n. Not applicable.

2. With respect to the compartments being analyzed:

a.} b.} Compartment pressure and temperature were not analyzed for two reasons. First, as stated in our original response to NUREG 0803, no equipment necessary to achieve cold shutdown following a scram is located at the 522' level; therefore, compartment conditions for equipment qualification is not a concern. Secondly, the integrated leak mass at 3.8 hours is only 9370 lbm which, even if held to the 522' compartment, would not significantly affect pressure or temperature. After 3.8 hours, any leakage (< 2 gpm) is at a temperature of less than 200oF and has negligible effect.

3. All assumptions used, including but not limited to the:

a. Orifice Coefficient: Moody critical flow was assumed out to 3.8 hours for an area equivalent to no seals and 10 gpm initial flow. For the hydrostatic pressure difference analysis, an orifice equation was used with a discharge coefficient of 0.7.

b. Fluid expansion factor: Not applicable.

NRC QUESTION 10.66, PART II

II. Verify that all electrical and mechanical equipment needed to mitigate the event is qualified to the environmental conditions determined in Part I. Verify that no pump cavitation will occur when pumping 212oF water.

Response to Part II

As stated in Part I.2 above, no equipment necessary for safe shutdown following scram is located in this area, therefore, environmental qualification is not an issue. The Supply System interprets ". . . no

pump cavitation . . ." to refer to the sump pump to which the reactor leakage drains (see Figure 1). If the total leakage (assuming no flashing) drains to the sump, its water level would be 0.2". Therefore, pump cavitation is not an issue as the sump pumps are not required to prevent flooding.

NRC QUESTION 10.66, PART III

III. Provide a discussion of the procedural steps to be taken to isolate the break in the CRD pipe at the outside surface penetration of containment to terminate the small pipe LOCA accident. Identify all equipment and materials required to isolate the break. Provide a commitment to maintain on site these items as dedicated equipment and materials. Discuss your procedure to verify periodically the existence and condition of these dedicated items.

Response to Part III:

After 3.8 hours the leak is reduced to $\leq$1.7 gpm and 200oF water from the severed withdrawal line. It should be noted that a leak of 1.7 gpm is possible only by neglecting all withdrawal line losses and assuming no CRD piston seals exist. The realistic flow rate would be much less. A temporary fix could be effected easily by plugging or freeze sealing the tube. Since a permanent fix becomes a question of availability and the total scenario given in Part I indicates no safety questions pertain to the incident, the Supply System does not feel a commitment to maintain and verify the existence of the equipment necessary to fix a leak of less than 1.7 gpm is prudent nor necessary.

NRC QUESTION 10.66, PART IV:

IV. Assuming that the sumps are inoperative for the event in Part I (since they are not seismic Category I, Class 1E), provide maximum water level in the compartment. Verify that no equipment required to bring the plant to a safe condition will fail as the result of internal flooding. Verify that no personnel radiation hazard will exist due to wading through reactor coolant.

Response to Part IV:

See Response to Part II of this question regarding water level and equipment necessary to achieve a cold shutdown. With the leak at <2 gpm and floor drains available, no personnel will be required to 'wade' through reactor coolant. As was true for Part III of this question, it is a matter of plant availability, not public health and safety, as to when personnel would enter. Access to the area can be controlled since no time constraints are imposed for stopping a leak of this magnitude.

Fire Area	Elevation	Total Fire Loading BTU/FT² [Including Transient (3)]	No. of Safety Related Cables Protected	Type of Combustibles: Lbs. of Cable Insulation	Type of Combustibles: Gallons of Oil/Lbs. of Grease/Cu.Ft. Charcoal	Remarks (6)	Response Time From Alarm Within:
RADWASTE & CONTROL BUILDING							
CABLE SPREADING ROOM (RC-II)	448'-0"	78,500	219 (1)	50,000	-	-	2 Min.
CABLE CHASE (RC-III)	467'-0" to 525'-0"	182,300	170 (1)	10,292	-	-	2 Min.
REMOTE SHUTDOWN ROOM (RC-IX)	467'-0"	45,000	32 (2)(4)	1,507	-	-	2 Min.
SWITCHGEAR ROOM#1 (RC-XIV)	467'-0"	29,620	2 (2)	2,054	-	-	2 Min.
TURBINE GENERATOR BUILDING							
CORRIDOR (TG-1)	441'-0"	75,780	26 (1)	7,808	-	-	3 Min.
REACTOR BUILDING							
GENERAL FLOOR AREA (R-I)	471'-0"	23,260		17,218	2.60 Gal.	Oil in Electro-Hydr. Operators	2 Min.
	501'-0"	29,230		14,762	-	-	2 Min.
	522'-0"	33,320	153 (2)	19,365	220.00 Gal.	Two RRC Hydr. Fluid Reservoirs, Separated and Curbed	2 Min.
	548'-0"	7,600		4,230	5.40 Gal.	Oil in Pump Motor & Electro-Hydr. Operators	3 Min.
	572'-0"	23,613		8,438	16.50 Gal. 57.5 Lbs. 179 Cu. Ft.	Oil in Electro-Hydr. Operators Grease in Valve Operators Charcoal in SGT Units (5)	3 Min.

NOTES:

(1) Cable raceway systems protected by 1-hour rated fire barrier/envelopes, automatic sprinkler protection, ionization-type smoke detectors, and manual hose/standpipe systems.

(2) Cable raceway systems protected by 3-hour rated fire barrier/envelopes, ionization-type smoke detectors, and manual hose/standpipe systems.

(3) A transient 55-gallon drum of lubricating oil was considered in all areas.

(4) Eight cables currently identified as requiring fire barrier/envelope protection; 24 cables under analysis.

(5) Protected by fixed water spray fire suppression system.

(6) For further details refer to WNP-2 FSAR, App. F, Fire Protection Evaluation.