Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

November 15, 1977

Mr. Donald K. Davis, Acting Chief
Operating Reactors - Branch 2
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Response to Fire Protection Questions
NRC Docket Nos. 50-237/249

Dear Mr. Davis:

Attached are three (3) copies of answers to the NRC questions submitted to Commonwealth Edison Company informally at the end of May, 1977, relative to Fire Protection at Dresden Station.

The following responses to NRC positions and questions are included in the Attachments:

Dresden Units 2 and 3

PF 15
PF 18
PF 20
PF 25

Please refer any further questions you might have to this office.

Very truly yours,

M.S. Turbak

M. S. Turbak
Nuclear Licensing Administrator
Boiling Water Reactors

Attachments

Dresden 2 & 3

POSITION

PF 15 Fire Protection Systems: Divisional Common Areas

Is the cable pan routing shown on Dresden 2 & 3 fire protection drawing 2/3.2-1-3 correct?

ANSWER

The cable pan routing shown on drawing 2/3.2-1-3 was compared to the routings shown on the electrical installation drawings and found to be correct.

POSITION:

PF.18

Identification

PF.18 - Fire Protection: Containment

Reference

Section F.(1) of Appendix A to BTP 9.5.1 and licensee's submittal of October 1, 1976, paragraph IV.D.1(a), page 1-15.

Licensee's Proposal

The licensee does not provide a fire detection system for the primary containments. For these units, the primary containment is inerted during normal operation. During refueling the inert is lifted to perform refueling and maintenance tasks.

Staff Concern

If a fire occurs during refueling and maintenance operation when it is most likely to occur, the fire could do significant damage to safety-related cabling or other safety-related equipment inside containment.

Position

Hose stations should be provided inside containment. A fire detection system should be provided inside containment.

Schedule

This position should be implemented within one year.

ANSWER:

Attached is the fire hazards analysis for a deinerted drywell at Dresden 2 & 3. This analysis supersedes the analyses for fire areas 1.2.1 (Unit 3 drywell) and 1.2.2 (Unit 2 drywell) upon installation of equipment to permit reactor operation with a deinerted containment. It also incorporates information contained in the June 27, 1977 response to position PF18.

2.3.1.2.1 Primary Containment - Unit 3

2.3.1.2.1.1 Fire Barrier Description

The drywell has structural reinforced concrete basemat, walls, and top plug. The elements vary in thickness from a 60 inch minimum thickness for the walls, to 72 inches for the top plug, and 18 feet 0 inch for the basemat. In the west wall from elevation 592 feet 6 inches to the top plug there are 48 inch thick steel plate lined concrete plugs for access to the spent fuel pool. In the east wall from elevation 588 feet 8 1/2 inches to the top plug there are 54 inch thick steel plate lined concrete plugs for access to the dryer and separator pool. At elevation 517 feet 6 inches there is an 11 foot 9 inch diameter equipment hatch with a 60-inch thick removable concrete shield plug door. Also at elevation 517 feet 6 inches there is a 10 foot 9 inch diameter personnel air lock. The walls and ceiling of the personnel airlock are 18 inch thick concrete. The floor is 30 inch thick concrete. The reactor shield plug at elevation 613 feet 4 inches is the roof of the drywell. It consists of three 24 inch thick steel lined and reinforced concrete slabs each having a minimum 4 1/2 inch bearing surface all around.

The floor of the drywell at elevation 502 feet 4 inches consists of an 8-inch separate concrete finish on 17 foot 4 inch structural concrete layers with a steelplate lining. Located at this level are 2 sump pits, a floor drain sump and an equipment drain sump. There are eight 6 foot 9 1/2 inch diameter vent lines at the base level. These lines vent to the suppression chamber.

The primary containment electrical penetrations are typical double sealed (one at each end) testable penetrations, designed to meet ASME Boiler and Pressure Vessel Code, Section III, Class B vessel requirements. The pressure seal at each end of the penetration consists of a steel header plate, glass fiber, glass cloth, epoxy sealant, potting compound and potting boards. The drywell purge system penetrates drywell walls. Eighteen inch diameter 3/8 inch thick carbon steel pipe is used in place of normal duct work in the drywell purge system. Double butterfly isolation valves are located in the piping outside containment and are normally closed hence there is no need for additional fire dampers.

2.3.1.2.2.2 Safety-Related, Reactor Shutdown and Cooling Equipment

Control Rod Drives and Piping
Target Rock Valve 3-0203-3A
Electromatic Relief Valves

- 0203-3B
- 0203-3C
- 0203-3D
- 0203-3E

Recirculation Loop Valves and Piping

MO-0202-4A
MO-0202-4B
MO-0202-5A
MO-0202-5B

Shutdown Cooling Valves and Piping

MO-1001-1A
MO-1001-1B

Standby Liquid Control Piping

SRM Drives

IRM Drives

Containment Isolation Valves

MO-1301-1 (Isolation Condenser)
MO-1301-4 (Isolation Condenser)
MO-2301-4 (HPCI)
AO-203-1A, 1B, 1C, 1D (Main Steam)

Reactor Coolant Pressure Boundary Valves and Piping

Safety Related Electrical Cables for the Neutron Monitoring System, Automatic Depressurization System (ADS), and Containment Isolation System

Safety Related Instruments

2.3.1.2.1.3 Protection Criteria and Measures

Fire protection requirements for the primary containment were recognized in the plant design. Reactor Protection System (RPS) cables are enclosed in metal conduit throughout the plant.

GE Design Criteria 22A2501 "Separation Requirements for Reactor Safety and Engineered Safeguards Systems" was followed in the installation of electrical cable in the drywell. Sections 4.3.1 and 4.3.2 have the following general criteria which apply to the RPS sensor wiring in the drywell and reactor vessel.

(1) Wiring from duplicate sensors connected to a common process tap should run in separate conduits to their separate destinations in order to meet the single failure criteria.

(2) Wiring for sensors of more than one variable in the same instrument channel may be run in the same conduit.

(3) Neutron monitoring cables routed through drywell penetrations shall be so separated that loss of all cabling in a single penetration cannot prevent a reactor scram.

The Dresden 2 & 3 FSAR states in Section 7.7.1.3 that, "RPS circuits are isolated to preclude a fault in one circuit from propagating to another and to reduce the likelihood that severe environmental influence, which might adversely affect reliability, will encompass more that one circuit. The sensors are dispersed; both sensors in one logic channel are not allowed to occupy the same general region or to be connected to a common header or point. The wiring is dispersed so that a single fire or accident would, most probably, not affect more than one input to the protection system."

For Primary Containment Isolation System (PCIS) and Engineered Safeguards Subsystems (ESS) the GE criteria requires that "in rooms containing high pressure feedwater piping or high pressure steam lines such as exist in the drywell, a minimum separation of 20 feet or a 6 inch thick reinforced concrete wall (or equivalent) is required between trays containing cables of different divisions. In making field runs of cable and conduit in the drywell, it may be impractical to meet this requirement in some instances. In cases where the above requirements cannot be met, the proposed design layout must be reviewed by Domestic Turnkey Project Engineering to assure adequacy of separation."

In the primary containment, ESS instrument cables are completely enclosed in solid steel cable trays and/or conduits.

All electrical cable penetrations are located between elevation 517 feet 6 inches and 545 feet 6 inches. Division I cable enter the drywell through two sets of penetrations east of the north-south reactor center. One set of penetrations is in the north-east quadrant. The other set is located in the southeast quadrant. Division II cable enter through penetrations west of the north-south reactor center line. One set in the northwest quadrant; the other in the southwest quadrant. Divisional separation is generally maintained in the drywell with Division I cable and equipment east of reactor center line and Division II cable and equipment west.

Mechanical equipment in the drywell is also physically separated such that a single credible fire would not disable all redundant and diverse systems that could be used for safe shutdown.

Shutdown reactor cooling (SRC) isolation valves MO-1001-1A (Loop A) and MO-1001-1B (Loop B) are located 130° apart on opposite sides of the reactor. Electromatic relief valves 0203-3C and 0203-3D in the auto depressurization system (ADS) are on the side opposite the reactor from target rock valve 0203-3A and electromatic relief valves 0203-3B and 0203-3E. Reactor recirculation loop A shutoff valves MO-0202-4A and MO-0202-5A and pump 3A-0202 are separated from loop B valves MO-0202-4B and MO-0202-5B and pump 3B-0202 by 180° with the reactor support pedestal between them.

Managerial procedures and controls are enforced during refueling and maintenance operations to control any additional hazards that could be introduced to the primary containment area. During unit outages that require entry to the containment, Maintenance Department procedure DMP 508 entitled "Fire Prevention Procedure For Welding and Cutting" is followed. This procedure makes clear the need and methods for fire prevention. In addition, operating personnel are on site on a 24 hour basis and as a part of their job make periodic rounds in the vicinity of the entrance of the drywell.

2.3.1.2.1.4 Combustible Material

See Table 2.3.1.2.1-1

2.3.1.2.1.5 Fire Loading

See Table 2.3.1.2.1-1

2.3.1.2.1.6 Extinguishing and Detecting Capabilities

Four CO_2 and dry chemical extinguishers and a hose station are located near drywell entrances to provide fire extinguishing capability during unit outages. Additionally, during a refueling outage or extended maintenance outage clean demineralized water is opened to the drywell and there are various hose stations inside to connect hoses.

TABLE 2.3.1.2.1-1

COMBUSTIBLE MATERIAL IN THE DRYWELL

Material	Quantity	NFPA Class	Location	BTU Content	Area[1]	Fire Load	Comment
Lubricating Oil	50 gal[2]	B	Reactor Recirculation Pump A	145,000 $\frac{BTU}{gal}$	1100 ft^2	6590 $\frac{BTU}{ft^2}$	
	50 gal[2]	B	Reactor Recirculation Pump B	145,000 $\frac{BTU}{gal}$	1100 ft^2	6590 $\frac{BTU}{ft^2}$	
HVAC Filters (Glass Fiber Prefilters)	112 elements	A	Drywell Coolers	1500 $\frac{BTU}{element}$	1100 ft^2	153 $\frac{BTU}{ft^2}$	There are 7 drywell coolers with 16 elements each
Cable[3] Insulation							
Cross-Linked Polyethylene (insulation)	28 ft^3	A	Throughout area	480,000 $\frac{BTU}{ft^3}$	1100 ft^2	12,220 $\frac{BTU}{ft^2}$	
Polyvinyl Chloride (jacket)	29 ft^3	A	Throughout area	693,000 $\frac{BTU}{ft^3}$	1100 ft^2	18,270 $\frac{BTU}{ft^2}$	Ignition point 850° F

Table 2.3.1.2.1-1 (cont.)

NOTES:

(1) The total drywell floor area surrounding the reactor support pedestal at elevation 502 feet 4 inches is 1100 ft^2.

(2) Only oil from one pump is assumed to be involved in a fire because of:

a. the 180° separation between the pumps with the reactor pedestal providing a barrier in the direct line between them,

b. the availability of drywell floor drains and sumps directly below each pump (see Figure 1). Four 4 inch diameter floor drains are located approximately in the center of each quadrant. These drains are connected to a 4 inch diameter pipe discharging directly into a sump. The gradient of the pipe establishes the maximum discharge capacity of the system at 100 gpm. The system does not have an oil separator. Should the two 50 gpm sump pumps fail the sump alone has a 942 gallon storage capacity. Oil would flow into these drains before completely covering the floor.

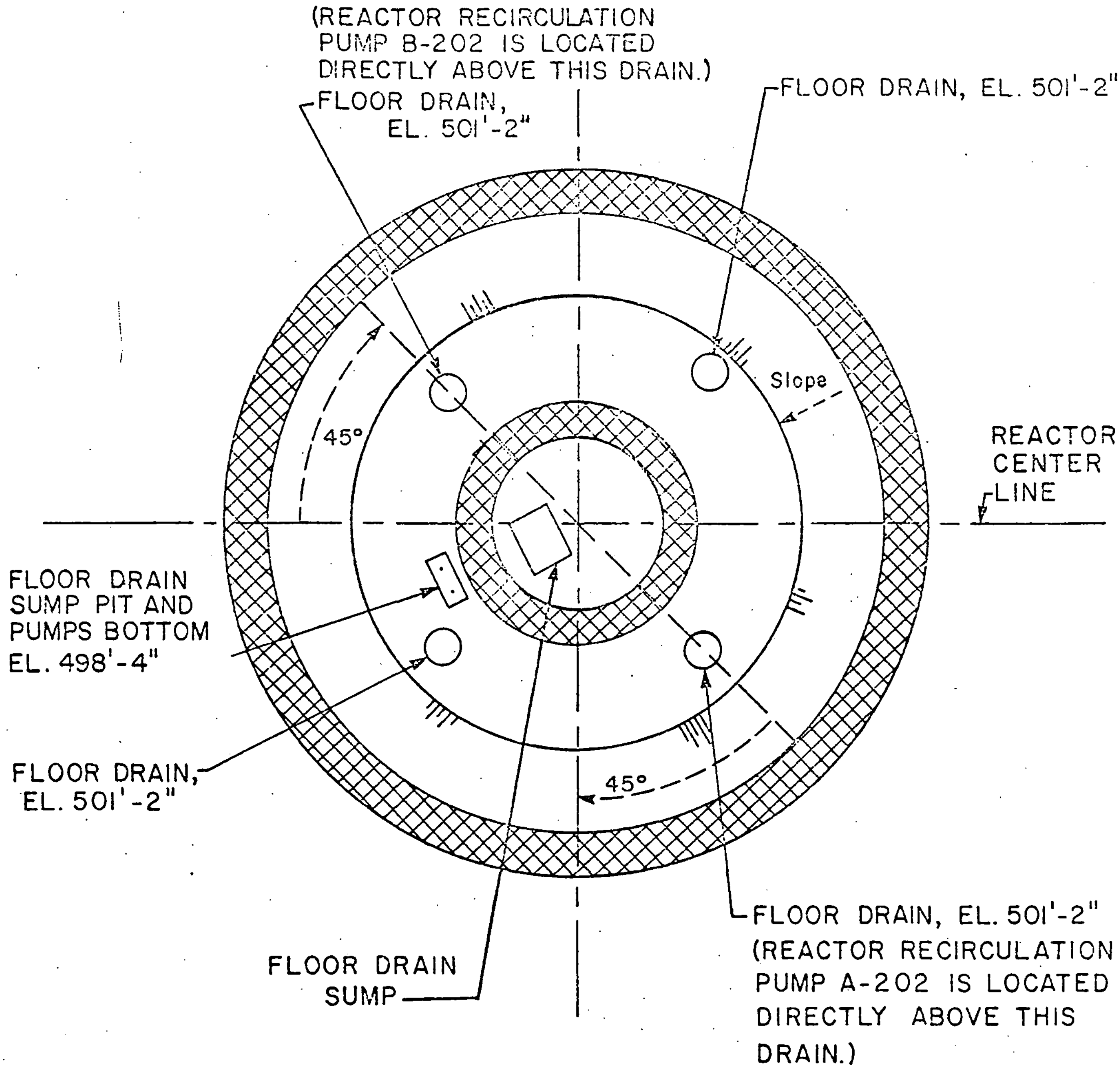
(3) The fuel contribution of cable insulation enclosed in conduit was neglected in the calculation of the fire load.

2.3.1.2.1.7 Design Basis Fire

A. Reactor Operation

In the event of a design basis fire, (as defined in BTP APCSB 9.5-1), the reactor could scram due to high drywell differential pressure signal (2 psig) or Average Power Range Monitor (APRM) channel inoperative signal and hot shutdown could be achieved using the isolation condenser and the HPCI and ADS systems for core cooling. The feedwater system will also still be available to provide makeup water to the core. The air operated main steam isolation valves, if they fail, will fail closed.

However, we do not consider a design basis fire to be a credible event. The largest concentrated source of combustible material is the 50 gallons of lubricating oil in the recirculation pumps. These pumps are separated from each other as described in Section 2.3.1.3.1.3. Drywell floor drains are placed at locations directly below each pump and at points midway between the pumps, thus preventing the spread of oil completely around the circumference of the drywell floor and affecting the other recirculation pump (see Figure 1). Redundant ADS valves are separated as described in Section 2.3.1.2.1.3. The SRC isolation valves must open for



DRESDEN STATION
Units 2 & 3

FIGURE 1

DRYWELL FLOOR DRAIN LOCATIONS

the system to function. These valves are also on opposite sides of the reactor near the recirculation pumps and only one is expected to be disabled by a credible fire. The loss of one loop of the SRC while in the shutdown mode will have no effect after the first 24 hours as that loop may be valved off without loss of shutdown capability. If the loss of a loop should occur before this time the reactor can still be cooled but not to 125° F in 24 hours. This system is not needed for hot shutdown.

Reactor vessel water level indication is needed for safe shutdown. All vessel water level instruments are located outside the drywell in separate instrument racks. Separate sensing lines each containing a condensing chamber are located on opposite sides of the reactor. Two instruments are located on each side of the reactor outside containment.

The drywell high pressure sensors will cause an alarm if the pressure exceeds 1.5 psig. Reactor scram, closure of drywell, vent, purge, and sump isolation valves, Traversing Incore Probe (TIP) withdrawal, standby gas treatment system (SGTS) activation and reactor building main vent isolation will occur at 2.0 psig.

Each Average Power Range Monitor (APRM) channel provides a scram trip signal to RPS if that channel is inoperative. If a Local Power Range Monitor (LPRM), used to provide input to an APRM channel, fails the operator can manually bypass this invalid input. If the number of bypassed LPRM's used as inputs to an APRM exceeds a preset number, the APRM instrument inoperative alarm is actuated. Should sufficient LPRM inputs be disabled in both RPS logic channels a scram would occur, thus causing reactor shutdown.

Control rod drive piping passes through the drywell. Because of the unique design of the locking piston control rod drive, an automatic scram occurs if both drive lines or only the outlet line is severed at any point with the reactor at pressure.

The standby liquid control system is provided to bring the reactor to a shutdown condition independent of the control rod system. This system has no active components in the drywell that could fail during a fire so as to disable the system.

B. Maintenance Operation

The effects of a fire in the drywell during a refueling or maintenance operation would be inconsequential. The Source Range Monitor (SRM) system would fail safe. Each of the four SRM channels initiates a rod block with the mode switch in either "Start up" or "Refuel", if a channel is inoperative.

The inoperative rod block is effective whenever the high voltage supply drops below a preset level, one of the channel modules is not plugged in, or the channel is not in its "operate" mode.

The shutdown reactor cooling water enters the drywell through LPCI piping and discharges into the reactor recirculation piping from which it enters the reactor. It takes suction from the reactor recirculation system. Each of two loops has a locked open gate valve and a testable air-operated check valve in the drywell prior to discharge and a motor-operated valve on the suction side. The reactor recirculation system is isolated from the shutdown reactor cooling system by two closed motor operated valves. The testable check valves are designed so that during operation, pump pressure will cause the disc to open regardless of the position of the air cylinder. No shutdown reactor cooling instrumentation is located in the drywell.

C. Radioactivity and Smoke Removal

Radioactivity released in the event of a fire during operation or refueling is not expected to exceed the amount that would be released during a drywell purge during normal plant operation.

The drywell and torus purge system may be utilized to remove the products of combustion from the drywell. The system is designed for approximately three air changes per hour in the drywell and torus which is sufficient for removal of the products of combustion. All equipment associated with this system is located outside the drywell.

2.3.1.2.1.8 Planned Modifications

None

2.3.1.2.2 Primary Containment - Unit 2

2.3.1.2.2.1 Fire Barrier Description

See Section 2.3.1.2.1.1

2.3.1.2.2.2 Safety - Related, Reactor Shutdown and Cooling Equipment

See Section 2.3.1.2.1.2

2.3.1.2.2.3 Protection Criteria and Measures

See Section 2.3.1.2.1.3

2.3.1.2.2.4 Combustible Material

See Section 2.3.1.2.1.4

2.3.1.2.2.5 Fire Loading

See Section 2.3.1.2.1.5

2.3.1.2.2.6 Extinguishing and Detecting Capabilities

See Section 2.3.1.2.1.6

2.3.1.2.2.7 Design Basis Fire

See Section 2.3.1.2.1.7

2.3.1.2.2.8 Planned Modifications

See Section 2.3.1.2.2.8

POSITION:

PF.20

Identification

PF.20 - Fire Pump Capacity

Reference

E.2.(c) of Appendix A to BTP 9.5.1 and licensee's submittal dated October 1, 1976, paragraph (e), page 1-12.

Licensee's Proposal

The licensee does not address the ability of the fire pumps to meet the maximum water flow demand with one pump inoperative. If credit were not given for the service water pumps this design would be inadequate.

Staff Concern

From the information obtained from the licensee during the site visit the existing fire pumps can not meet the maximum water demand with the one pump out of service.

Position

The licensee should provide a sufficient number of pumps so that 100% capacity will be available with one pump out of service. The pump installation should conform to the requirements of NFPA 20.

Schedule

This position shall be implemented within one year.

ANSWER:

The existing fire pump capacities are: -

Unit	Pump	Number	Capacity (gpm)	Pressure (psig)
1	Jockey Fire Pump	1	50	100
1	Screen Wash Pump	2	2000	62.5

ANSWER: PF.20 (Cont'd)

Unit	Pump	Number	Capacity (gpm)	Pressure (psig)
2/3	Service Water Pump	4*	15,000	91
1	Diesel Criven Fire Pump	1	1,000	100
2/3	Diesel Driven Fire Pump	1	2,000	125

* Five service water pumps are available, but only four are normally needed in service at any one time.

Hydraulic calculations were performed for the largest sprinkler demand at Dresden 2 & 3 with only the Dresden 1 fire pump and the one Dresden 1 screen wash pump to meet the demand. The largest demand for Dresden 2 & 3 is the fire water sprinkler system in the turbine building mezzanine floor as described in the response to Dresden 2 & 3 Question 31. Three 1-1/2 inch hose reels are available in the area. A flow rate of 100 gpm was assumed for each hose reel. The analysis considered all available paths for water flow from the pumps in the Unit 1 crib house to the point of demand. In the network analysis, equivalent lengths of piping were established by converting all piping to an equivalent length having a Hazen Williams coefficient equal to 100 (C=100), and taking into account all fittings.

The flow and pressure requirements at the pumps in the Unit 1 crib house are 2232 gpm at 113 psi. The available supply with the Unit 1 fire pump and one screen wash pump is 2232 gpm at 102 psi. To postulate the multiple failures of both the Units 2 & 3 service water pumps, the Unit 2 & 3 fire pump and one unit 1 screen wash pump simultaneously with a fire involving all the sprinkler heads in the sprinkler with the largest water demand at Dresden is not credible. The available supply with the unit 1 fire pump and both screen wash pumps is 2232 gpm at approximately 120 psi. Additionally the Unit 1 screen wash pumps can be operated by onsite emergency power provided by the Dresden 1 diesel generator. This supply is sufficient to meet the demand. Copies of the hydraulic calculations are attached.

The results of calculations showing the ability of Dresden 2 & 3 fire pump to meet the largest demands at Dresden 1 and Dresden 2 & 3 with an impairment in the fire main in the worst location are discussed in the response to Dresden 2 & 3 Question 31.

REFERENCES:

1. Dresden 2 & 3 FSAR Amendments 7 (Unit 2) and 8 (Unit 2), question 9,9, page 9.9-1.

HYDRAULIC CALCULATIONS

FOR Commonwealth Edison
Dresden / Fire & Screen Wash Pumps

SHEET 1 OF 1
BY ALH
DATE 10/13/77

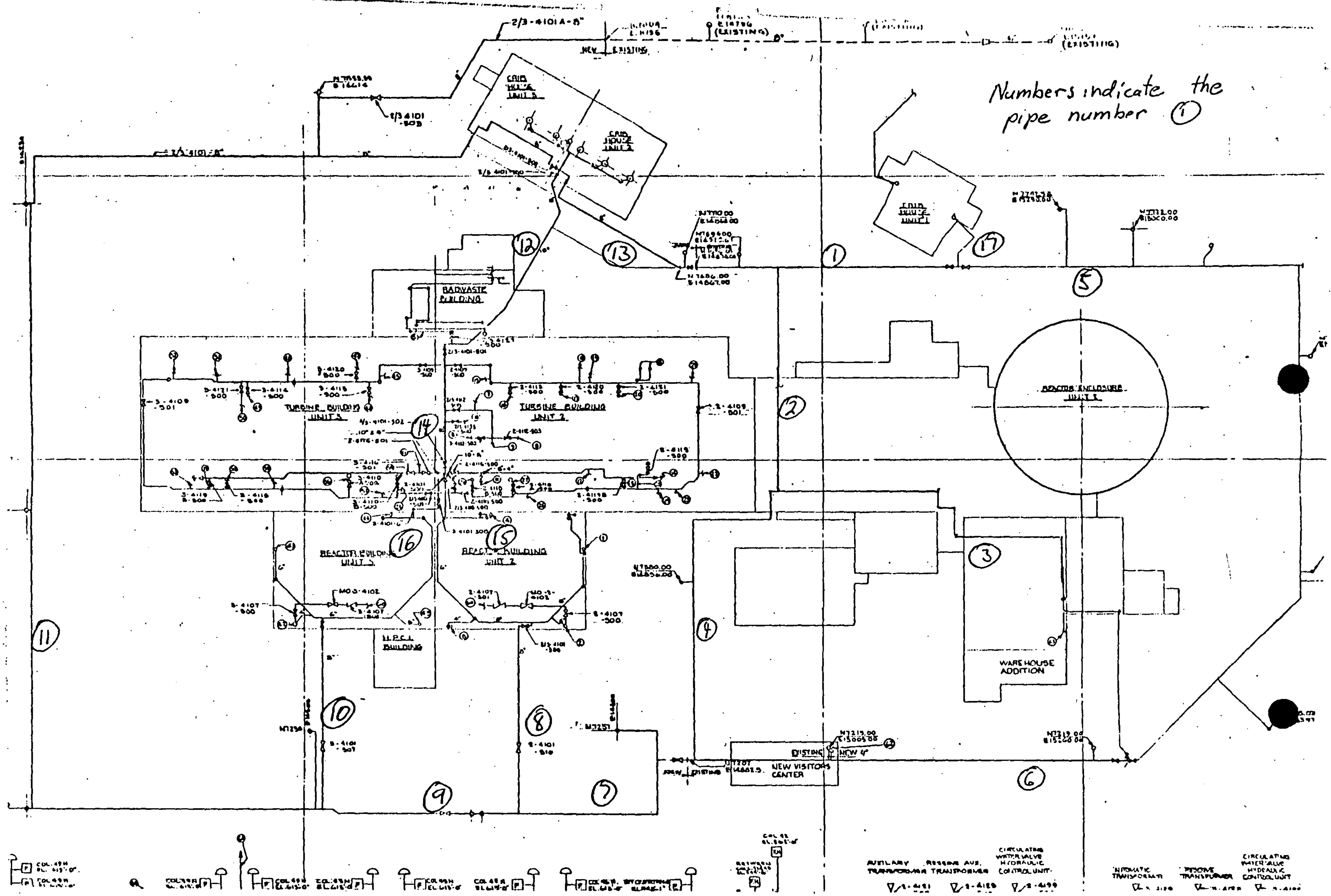
NOZZLE IDENT. AND LOCATION	FLOW IN G.P.M.	PIPE SIZE	PIPE FITTINGS AND DEVICES	EQUIV. PIPE LENGTH	FRICTION LOSS P.S.I./FOOT	PRESSURE SUMMARY	NORMAL PRESSURE	NOTES
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q 2232			T		Pf	Pn	
				L		Pt 61.0	Pt	DEMAND
	q		SEE COMPUTER	F		Pe	Pv	
	Q		PRINTOUT	T		Pf 31.0	Pn	
PIPE 17			1 EE	L 100	C=100	Pt 92.0	Pt	
	q	8	2 E	F 62	ELEV 38'-11"	Pe 12.52	Pv	
	Q 2232		1 T	T 162	.054	Pf 8.75	Pn 113.27	@ PUMP FLANGE
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
				L		Pt	Pt	
	q			F		Pe	Pv	
	Q			T		Pf	Pn	
						Pt		

DRESDEN NO. 1
JOB NO. -
ACCUR. - .100000
DELTA - .766132-01

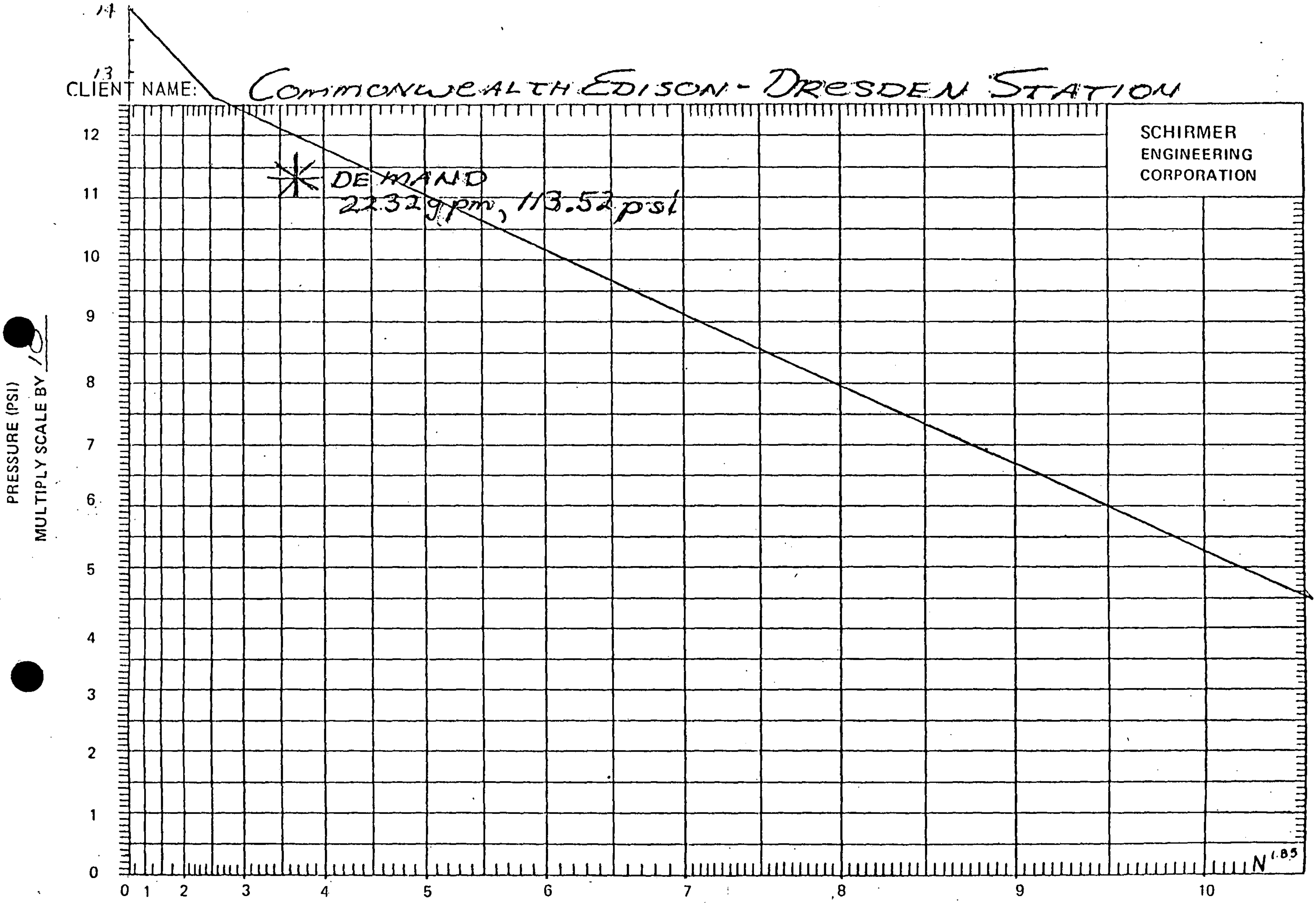
PIPE	H-LOSS	GPM	DIA	LENGTH	PLN	PSN
1	6.0	1646	8.070	195	3	0
2	-.2	-206	8.070	272	4	3
3	.5	187	8.070	328	2	3
4	1.0	393	8.070	446	2	4
5	-5.8	-685	8.070	942	3	0
6	-1.4	-498	8.070	420	2	0
7	-2.9	-891	8.070	287	4	0
8	.6	470	8.070	201	1	4
9	-.5	-422	8.070	212	1	0
10	-.4	-340	8.070	239	1	5
11	-.2	-81	8.070	1798	5	0
12	2.3	1521	10.190	269	4	5
13	4.8	1440	8.070	197	4	0
14	-.1	-810	10.190	54	4	5
15	2.3	470	6.080	191	1	4
16	-2.0	-340	6.080	297	1	5

21 ITERATIONS

31.0



Numbers indicate the pipe number (1)
CRIB HOUSE UNIT 3
CRIB HOUSE UNIT 2
CRIB HOUSE UNIT 1
RADWASTE BUILDING
TURBINE BUILDING UNIT 3
TURBINE BUILDING UNIT 2
REACTOR ENCLOSURE UNIT 1
REACTOR BUILDING UNIT 3
REACTOR BUILDING UNIT 2
H.P.C.I. BUILDING
WAREHOUSE ADDITION
NEW VISITORS CENTER
EXISTING
NEW 4"
2/3-4101A-8"
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17



MULTIPLY SCALE BY 600

FLC - GPM

COMBINED CURVE
2 SCREEN WASH PUMP -
(2000 gpm, 62.5 psi)
1 FIRE pump (1000 gpm, 100 psi)

MGU8-1

Position:

PF25

Identification

PF.25 - Radiological Consequences of Fires

Reference

Section A.(2) of Appendix A to BTP 9.5-1 and licensees submittal of March 29, 1977. Section 2.3.14.1.7 on page 2.3-72.

Staff Concern

The licensee has not evaluated the radiological consequences of fires which may involve radioactive material, such as fires in the radwaste areas (e.g., involving dried radioactive filter sludges or other dry radioactive materials) or the off-gas system. Thus reasonable assurance has not been provided that the fire protection systems associated with such fires are adequate to minimize the potential for radiation releases.

Staff Position

The licensee should analyze the radiological consequences of fires in the radwaste areas and off-gas systems. The consequences of fires should not significantly exceed these previous calculated for accidents involving radwaste systems.

Answer

Within the radwaste system there are dry active materials (such as paper, rags, etc.), located in the dry active waste compactor and storage area of the radwaste building. These are the only combustible radioactive materials in the radwaste system and because of the low level activity of this material a fire would not produce radiological release consequences exceeding 10CFR20 limits. All other radwaste material is contained in tanks with water or in 55-gallon drums solidified in cement. It is inconceivable to postulate a fire involving this material.

The potential radiological consequences of the Dresden Units 2 & 3 off-gas system component failure have been submitted to the Atomic Energy Commission (now the Nuclear Regulatory Commission) in Table 4 of Quad Cities Units 1 & 2 Special Report No. 1 and Supplementary Information for Dresden Units 2 & 3 Special Report No. 4A, Modified Off-Gas System, Commonwealth Edison Company, June 11, 1974. The docket number is 50237-276. No credible hypothetical fires in the off-gas system could produce doses to the public above those calculated, using very conservative assumptions, in this report.