ATTACHMENT 6

Number of each class of safety related cables routed in fire zone.

Plant Usage	Number	Safety Classification	Channel or Division*
Common	20	Engineered Safeguard - ECCS	I
	20	Engineered Safeguard - ECCS	II
	13	Engineered Safeguard - Diesel A	IA
	33	Engineered Safeguard - Diesel C	IIC
	5	Engineered Safeguard - Diesel D	IID
	7	Load Shedding - Diesel A	A1
	9	Load Shedding - Diesel C	B1
	7	Supporting Auxiliaries - Electrical	IE
Subtotal	114		
Unit 1	6	Engineered safeguard - ECCS	I
	182	Engineered Safeguard - ECCS	II
	4	Load Shedding - Diesel A	A1
	5	Load Shedding - Diesel C	B1
	1	Load Shedding - Diesel D	B2
	52	Neutron Monitoring (also activates	IA
	52	Neutron Monitoring RPS)	IB
	52	Neutron Monitoring ↓	IIA
	52	Neutron Monitoring ↓	IIB
	14	Primary Containment Isolation	I
	39	Primary Containment Isolation	II
	2	Reactor Protection (control rod	IA
	2	Reactor Protection scram)	IB
	2	Reactor Protection ↓	IIA
	2	Reactor Protection ↓	IIB
	3	Reactor Protection ↓	IIIB
	12	Supporting Auxiliaries - Electrical	IE
Subtotal	482		
Unit 2	15	Engineered Safeguard - ECCS	I
	3	Engineered Safeguard - ECCS	II
	4	Supporting Auxiliaries - Electrical	IE
Subtotal	22		
Unit 3	4	Engineered Safeguards - ECCS	I
	3	Engineered Safeguards - ECCS	II
	3	Supporting Auxiliaries - Electrical	IE
Subtotal	10		
TOTAL	628		

*See Attachment 5 for channel or division definitions.

ATTACHMENT 7 SHEET 1

Summary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	INSULATED*	NO. CABLES DAMAGED	CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	INSULATION*	NO. CABLES DAMAGED
WBB	1/c # 12	1	11+	WLB	2/c # 12	4	[illegible]
WCA	1/c # 14	1	2+	WLC	3/c # 12	4	[illegible]
WDD	1/c # 8	2	4+	WLG	7/c # 12	4	[illegible]
WDE	1/c # 6	2	1+	WLN	2/c # 10	4	[illegible]
WDF	1/c # 4	2	7+	WLO	3/c # 10	4	[illegible]
WDG	1/c # 2	2	13+	WLS	7/c # 10	4	[illegible]
WDH	1/c #1/0	2	2+	WNB	1/c #2/0	5	[illegible]
WDI	1/c #2/0	2	2+	WNC	1/c #4/0	5	[illegible]
WDK	1/c #4/0	2	2+	WNF	1/c #500	5	[illegible]
WDN	1/c #300	2	6+	WTA	50 pr #19	6	[illegible]
WDO	1/c #400	2	1+	WTD	12 pr #19	6	[illegible]
WFB	2/c # 10	3	16	WTJ	COAX	7	[illegible]
WFC	3/c # 10	3	1	WTK	COAX	8	[illegible]
WFD	4/c # 10	3	6	WTK-1	COAX	9	[illegible]
WFE	5/c # 10	3	1	WTK-2	COAX	10	[illegible]
WGB	2/c # 12	3	157	WTO	2 pr #18	11	[illegible]
WGC	3/c # 12	3	18	WTR	2 pr #14	12	[illegible]
WGD	4/c # 12	3	37	WUB	Thermocouple	13	[illegible]
WGE	5/c # 12	3	13	WUB-1	Thermocouple	14	[illegible]
WGG	7/c # 12	3	14	WUA	2/c # 16	15	[illegible]
WGI	9/c # 12	3	45	WVA-1	2/c # 18	15	[illegible]
WGK	12/c # 12	3	18	WVB	3/c # 16	15	[illegible]
WGM	16/c # 12	3	2	WVC	4/c # 16	15	[illegible]
WGN	19/c # 12	3	2	WVE	7/c # 16	15	[illegible]
WHB	2/c # 14	3	62	WVG	12/c # 16	15	[illegible]
WHC	3/c # 14	3	15	WVI	27/c # 16	15	[illegible]
WHD	4/c # 14	3	18	WVJ	37/c # 16	15	[illegible]
WHE	5/c # 14	3	13	WVR	29/c # 20	16	[illegible]
WHG	7/c # 14	3	10	WVU	5/c # 18	16	[illegible]
WHI	9/c # 14	3	8	WVU-1	8/c # 18	16	[illegible]
WHJ	9/c # 14	3	7	BELDON 8213	COAX	7	[illegible]
WHL	16/c # 14	3	1	MFR'S TV CABLE (BELDON 8212)	COAX	7	[illegible]
WHT	7/c # 16	3	8				

* No. of individual cable designations. Total no. of conductors for each designation appear on Attachment 9 showing cross sections of cables on trays.

* Numbers listed correspond to insulation of cable type as shown below. Refer to standard TVA specifications(Att. 13 & 14)and list of requisitions for signal cable (Attachment 8).

1. Single conductor power or control cable with polyethelene insulation and a nylon jacket over the polyethelene. (Termed "PN" per TVA Specification).

2. Single conductor power or control cable with cross-linked polyethelene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethelene. (Termed "CPJ" per TVA Specification).

ATTACHMENT 7 SHEET 2

3. Multiple-conductor cable with a core of the specified number of single conductors as in 1 above covered by a polyvinyl-chloride outer jacket. (Termed "INJ" per TVA Specification).

4. Multiple-conductor cable with a core of the specified number of single conductors as in 2 above covered by a polyvinyl-chloride outer jacket. (Termed "CPJJ" per TVA Specification).

5. Single conductor high-voltage (5000 volts) power cable with extruded stand and cross-linked polyethene insulation with metallic electrostatic shielding and polyvinyl chloride jacket overall. (Termed "CPSJ" per TVA Specification).

6. Telephone cable with high density polyethelene over each conductor, mylar backed rubber cable tape, aluminum shield, and high density polyethelene jacket overall. Some of these had polyvinyl jacket overall.

7. Coaxial signal cable with both conductor and overall jacket insulated with polyethelene.

8. Coaxial signal cable with conductor insulated with polyethelene and polyvinyl chloride jacket overall.

9. Coaxial signal cable with conductor and overall jacket insulated with irradiated blend of polyolefins and polyethelene and noise free. Some of these types had cross-linked polyethelene over both.

10. Same as 9 but made noise free by a carbon suspension.

11. Same as 6 except without shield.

12. Twisted pair cable with polyethelene over each conductor and polyvinyl chloride jacket overall.

13. Thermocouple cable with high density polyethelene over each conductor, aluminum foil/mylar type laminated shield, and high density polyethelene overall.

14. Thermocouple cable with heat and light stabilized cross-linked polethelene over each conductor, aluminum foil/mylar tape shield, and chlorosulfonated polyethel jacket overall.

15. Signal cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape laminated shield, fiberglass reinforced silicone tape assembly wrap, and chlorosulfonated polyethelene jacket overall.

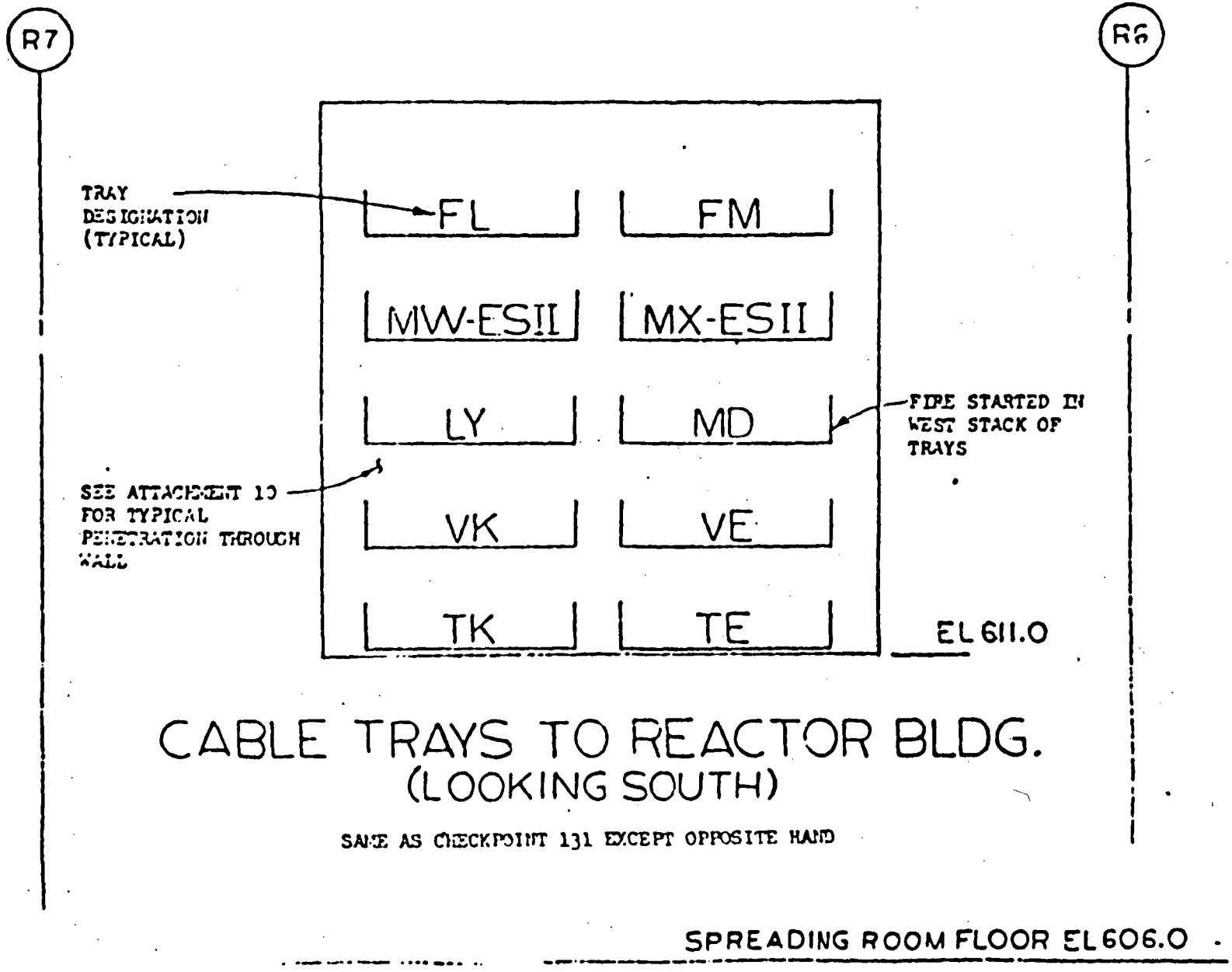
16. Multiple-conductor cable with core of specified number of single conductor cables insulated with cross-linked polyethelene and a neoprene jacket overall.

ATTACHMENT 8
SHEET 1
LIST OF REQUISITIONS FOR CABLE NOT PURCHASED
PER STANDARD TVA SPECIFICATION

CABLE TYPE(MARK)	REQUISITION NO.	REQUISITION ITEM	VENDORS
WTA	83451	1	Pwr, Telephone & Supply Company
	91972	1	Anaconda
WTD	75166	2	Anixter
	83451	4	Pwr, Telephone & Supply Company
	85202	3	Plastic Wire & Cable
	86712	1	Phalo Corporation
WTJ	61988	3	Okonite
	83470	2	Bourlurant Bros. Company
	84375	1	Dearborn
	84423	1	Alpha Wire
	85488	1	Belden
WTK	61737	4	Continental Wire & Cable
	61988	2	Alpha Wire
	84375	2	Dearborn Wire & Cable
	84453	1	Delco
	85488	2	Belden
WTK-1	61737	1	Brandel-Stephens Company
	75321	1	Boston Wire & Cable
WTK-2	75321	2	Boston Wire & Cable
WTO	83451	7	Pwr, Telephone & Supply Company
	91972	5	Sumitomo Shoji
WTR	64863	12	Jefferson Wire & Cable
	83451	8	Plastic Wire & Cable
WUB	61959	1	Continental Wire & Cable
WUB-1	54336	1	Continental Wire & Cable
	83427	1	Continental Wire & Cable
	85464	1	Continental Wire & Cable
WVA	54335	1	Boston Ins. Wire & Cable
	64863	1	Graybar Electric
	74910	13	Continental Wire & Cable
	83944	1	Continental Wire & Cable
WVA-1	54335	1	Boston Ins. Wire & Cable
	61986	1	Continental Wire & Cable
	74910	2	Continental Wire & Cable
	83944	1	Continental Wire & Cable

ATTACHMENT 8 SHEET 2

CABLE TYPE(MARK)	REQUISITION NO.	ITEM	VENDORS
WVB	54335	3	Boston Ins. Wire & Cable
	64863	2	Graybar Electric
	74910	3	Okonite Wire & Cable
	83849	1	Boston Ins. Wire & Cable
WVC	54335	4	Boston Ins. Wire & Cable
	64863	3	Graybar Electric
	74910	4	Okonite Wire & Cable
	83849	2	Boston Ins. Wire & Cable
WVI	54335	6	Boston Ins. Wire & Cable
	64863	9	Graybar Electric
	74910	10	Okonite Wire & Cable
WVJ	54335	7	Boston Ins. Wire & Cable
	64863	10	Graybar Electric
	74910	11	Okonite
	83849	7	Boston Ins. Wire & Cable
WVR	61844		Sies Electric Supply Company
	85529		ITT Surprenant
WVU	61985	1	General Electric
	75025	1	General Electric
	83426	1	General Electric
WVU-1	61985	2	General Electric
	75025	2	General Electric
	83426	2	General Electric
	85551	1	General Electric


R7
R6
TRAY DESIGNATION (TYPICAL)
FL
FM
MW-ESII
MX-ESII
LY
MD
FIRE STARTED IN WEST STACK OF TRAYS
SEE ATTACHMENT 10 FOR TYPICAL PENETRATION THROUGH WALL
VK
VE
TK
TE
EL 611.0
CABLE TRAYS TO REACTOR BLDG.
(LOOKING SOUTH)
SAME AS CHECKPOINT 131 EXCEPT OPPOSITE HAND
SPREADING ROOM FLOOR EL 606.0

(Looking South)

ATTACHMENT 9 SHEET 2

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
[illegible]	WDF	2	.429	.290
	WDG	12	.485	2.220
	WDK	3	.769	1.395
	WDM	3	.940	2.082
	WDO	4	1.029	3.332
	WLC	1	.539	.229
TOTAL		25		9.573
MX-ESII	WDD	2	.340	.182
	WDE	2	.379	.226
	WFB	1	.429	.145
	WFE	2	.659	.684
	WGB	58	.425	8.236
	WGC	1	.444	.155
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	4	.660	1.026
	WGI	37	.710	14.652
	WGK	3	.789	1.47
	WGM	1	.874	.600
	WHB	26	.384	3.016
	WHC	10	.405	1.290
	WHD	4	.439	.608
	WHE	7	.480	1.267
	WHG	4	.529	.848
	WHI	4	.640	1.288
	WHJ	4	.710	1.584
	WHL	1	.781	.479
	WVA-1	2	.439	.304
	WWN	1	.148	.0172
TOTAL		181		58.290
MD	WGB	1	.425	.142
	WGE	1	.559	.246
	WHB	1	.384	.116
	WHD	2	.439	.304
	WHE	1	.480	.181
	WTO	1	.340	.091
	WHR	1	.360	.102
	WVE	1	.461	.167

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FL	WDK	6	.940	4.164
	WLB	1	.509	.204
	WLH	1	.559	.245
	WLO	1	.627	.309
TOTAL		9		4.922
MW-ESII	WFB	9	.490	1.701
	WFD	1	.600	.283
	WGB	23	.425	3.266
	WGC	6	.444	.930
	WGD	3	.484	.552
	WGE	2	.559	.492
	WGI	3	.710	1.183
	WGK	3	.789	1.47
	WHB	14	.384	1.624
	WHE	5	.480	.905
	WHJ	1	.710	.396
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	Belden 8213	1	.405	.129
TOTAL		73		11.919
LY	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	3	.600	.849
	WGB	8	.425	1.136
	WGC	2	.444	.310
	WGD	2	.484	.368
	WGE	5	.559	1.230
	WGG	5	.660	1.710
	WGK	1	.789	.490
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHI	3	.640	.966
	WVA-1	2	.333	.174
TOTAL		41		5.783

ATTACHMENT 9 SHEET 3

Checkpoint 131
(Looking South)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
MD (Continued)	WVG	1	.587	.271
	WVI	11	.834	7.205
	WVJ	16	1.012	12.880
	WVR	101	.650	33.532
	WVU-1	3	.439	.456
TOTAL		141		55.693
VE	WVA	52	.353	5.096
	WVA-1	61	.333	5.307
	WVB	15	.371	1.620
	WVC	10	.401	1.260
	MFRS	1	.242	.046
TOTAL		139		13.329
TE	WUB	37	.231	1.974
	WUB-1	47	.339	4.230
	WVA	3	.353	.2940
TOTAL		87		6.498

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	41	.353	3.332
	WVA-1	46	.333	4.002
TOTAL		87		7.334
TK	WUB	34	.231	1.42[illegible]
	WUB-1	25	.339	2.250
	WVA	2	.353	.196
TOTAL		61		3.874

ATTACHMENT 9 SHEET 4

Checkpoint 102
(Looking North)

VE

TE

KS-ESII

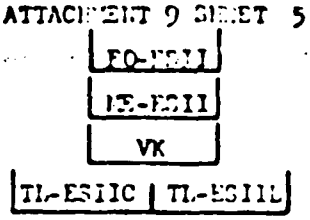
KT

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	67	.353	6.566
	WVA-1	38	.333	3.306
	WVB	3	.371	.324
	WVC	14	.401	1.764
	MFR	1	.242	.046
TOTAL		123		12.006
TE	WUB	117	.231	4.914
	WUB-1	25	.339	2.250
TOTAL		142		7.164

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
KS-ESII	WFB	1	.490	.18
	WFE	2	.659	.65
	WGB	40	.425	5.68
	WGC	3	.444	.46
	WGD	2	.484	.36
	WGE	1	.559	.24
	WGG	3	.660	1.02
	WGI	1	.710	.39
	WGK	1	.789	.49
	WHB	6	.384	.69
	WHC	2	.405	.25
	WHE	4	.480	.60
	WHG	1	.519	.21
	WHI	3	.640	.96
	WHJ	1	.710	.39
	WLB	1	.509	.20
TOTAL		72		12.0
KT	WGB	1	.425	.14
	WGG	1	.660	.34
	WHB	2	.384	.23
	WHD	7	.439	1.06
	WHG	1	.519	.21
	WTO	1	.340	.09
	WTR	1	.360	.10
	WVE	1	.461	.16
	WVG	1	.587	.27
	WVI	11	.834	7.20
	WVJ	8	1.012	6.44
	WVR	101	.650	33.53
	WVU-1	2	.439	.30
TOTAL		138		50.10

ATTACHMENT 9 SHEET 5

Checkpoint 127
(Looking North)

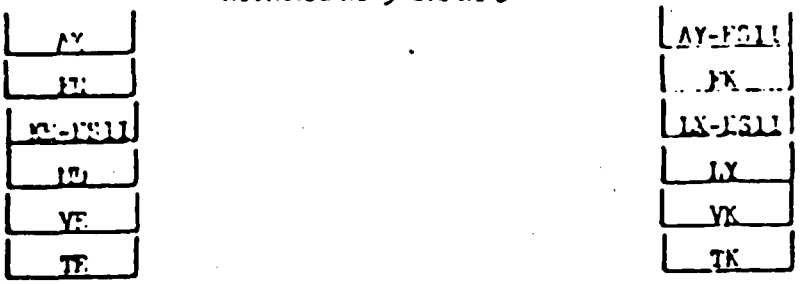


TRAY MISIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	4	.429	.580
	WDG	7	.485	1.295
	WDH	3	.619	.903
	WDI	3	.660	1.026
	WLB	1	.509	.204
	WLC	4	.539	.916
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		30		6.678
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFB	1	.490	.189
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU-1	2	.439	.304
	WWN	2	.0172	.0344
TOTAL		137		33.530

TRAY MISIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	8	.353	.78
	WVA-1	20	.333	1.74
	WVB	9	.371	.97
TOTAL		37		3.49
TL-ESIIC	WFB	8	.490	1.51
	WFD	1	.600	.28
	WGB	21	.425	.28
	WGC	5	.444	.77
	WGD	1	.484	.18
	WGE	2	.559	.49
	WGI	2	.710	.79
	WGK	3	.789	.14
	WHB	6	.384	.69
	WHD	1	.439	.15
	WHE	1	.480	.1
	WHG	4	.519	.8
	WTO	9	.340	.8
	WTR	1	.360	.10
	Belden 8213	1	.405	.12
TOTAL		70		7.39
TL-ESIIL	WUB	40	.231	1.6
	WUB-1	4	.339	.3
	WVA	8	.353	.78
TOTAL		52		2.8

Checkpoint 128
(Looking East)

ATTACHMENT 9 SHEET 6



TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	WHB	6	.915	3.942
FN	WLC	1	.539	.229
	WLS	1	.817	.524
	WDG	12	.485	2.220
	WDH	13	.940	9.022
	WGD	5	.484	.920
TOTAL		32		12.915
KE-ESII	WHE	2	.480	.362
	WHJ	3	.710	1.198
	WDD	2	.340	.182
	WFE	1	.659	.342
	WGB	4	.425	.568
	WGC	1	.660	.342
	WHB	22	.384	2.552
	WWH	1	.148	.0172
TOTAL		36		5.708
MD	WHD	7	.439	1.064
	WHE	1	.480	.181
	WHG	1	.519	.212
	WGB	9	.425	1.278
	WGE	1	.559	.246
	WGG	1	.660	.342
	WHB	2	.384	.232
	WVA-1	2	.333	.174
	WVI	1	.834	.655
	WVJ	8	1.012	5.240
	WVU-1	1	.439	.152
TOTAL		34		9.776

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0			
FK	WDF	8	.429	1.160
	WDG	10	.485	1.850
	WDH	5	.619	1.505
	WDI	3	.660	1.026
	WDK	1	.769	.465
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
TOTAL		41		15.268
LX-ESII	WHC	1	.405	.129
	WHE	1	.480	.181
	WTO	7	.340	.637
	WGC	1	.444	.155
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGG	1	.660	.342
	WGI	1	.710	.396
	WGK	2	.789	.980
TOTAL		17		3.496
LY	WTD	1	.638	.320
	WTO	24	.340	2.184
	WTR	2	.360	.204
	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	4	.600	1.132
	WGB	10	.425	1.420
	WGC	2	.444	.310
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGG	5	.660	1.710
TOTAL		56		8.819

ATTACHMENT 9 SHEET 7

Checkpoint 128
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	94	.353	9.212	LY	WGK	1	.789	.490
	WVA-1	85	.333	7.[illegible]95		WIB	6	.384	.696
	WVB	12	.317	1.296					
	WVC	4	.401	.504	TOTAL		7		1.186
TOTAL		195		18.407	VK	WVA	29	.353	2.842
						WVA-1	91	.333	7.917
						WVB	9	.371	.972
TE	WUB	80	.231	3.360					
	WUB-1	23	.339	2.070	TOTAL		129		11.731
	WVA	3	.353	.294					
					TK	WUB	16	.231	.672
TOTAL		106		5.724		WVA	6	.353	.588
					TOTAL		22		1.260

Checkpoint 129
(Looking East)

ATTACHMENT 9 SHEET 8

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX		0	0	0
FN		0	0	0
KE-ESII	WDD	2	.340	.182
	WFB	1	.490	.189
	WFE	1	.659	.342
	WGB	43	.425	6.106
	WGC	3	.444	.465
	WGD	6	.484	1.104
	WGE	1	.559	.246
	WGG	4	.660	1.368
	WGI	1	.710	.396
	WHB	26	.384	3.016
	WHC	2	.405	.258
	WHE	4	.480	.724
	WHG	2	.519	.424
	WHI	2	.640	.644
	WHJ	4	.710	1.584
	WLB	1	.509	.204
	WWN	1	.0172	.0172
TOTAL		104		18.024
ND		0	0	0
VE		0	0	0
TE		0	0	0

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0	0	0
FK-ESII	WDF	10	.429	1.45
	WDG	7	.485	1.29[illegible]
	WDH	2	.619	.60[illegible]
	WDK	4	.769	1.86[illegible]
	WDM	8	.940	5.55[illegible]
	WDO	4	1.029	3.33[illegible]
	WFB	2	.490	.37[illegible]
	WHB	1	.384	.11[illegible]
	WLB	1	.509	.20[illegible]
	WLM	1	.559	.24[illegible]
	WLO	2	.627	.6[illegible]
	WVA	2	.353	.1[illegible]
TOTAL		44		14.54[illegible]
IX-ESII	WDD	4		.364
	WDE	2		.226
	WFB	1	.490	.189
	WGB	25	.425	3.550
	WGD	3	.484	.552
	WGE	5	.559	1.230
	WGG	2	.660	.684
	WGI	36	.710	14.256
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.695
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	3	.519	.636
	WHI	4	.640	1.288
	WHJ	7	.710	2.872
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.000
	WVU-1	2	.439	.304
	WWN	2	.0172	.034
TOTAL		123		33.718

ATTACHMENT 9 SHEET 9

Checkpoint 129
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
					TK-ESIIC	WFB	8	.490	1.512
						WFD	1	.600	.283
						WGB	16	.425	2.272
						WGC	6	.444	.930
						WGD	2	.484	.363
						WGE	2	.559	.492
						WGI	2	.710	.792
						WGK	4	.789	1.960
						WHB	4	.384	.464
						WHC	1	.405	.129
						WHD	1	.439	.152
						WHE	2	.480	.362
						WHG	4	.519	.848
						WTO	1	.340	.091
						WTR	1	.360	.102
						Belden 8213	1	.405	.129
					TOTAL		56		10.886
					TK-ESIIL	WUB	1	.384	.116
						WUB	34	.231	1.428
						WUB-1	4	.339	.360
						WVA	4	.353	.392
					TOTAL		43		2.296

Checkpoint 130
(Looking West)

ATTACHMENT 9 SHEET 10

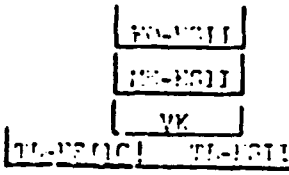
TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX		0		
FN	WDH	4	.940	2.576
	WLS	1	.817	.524
TOTAL		5		3.100
KE-ESII	WLB	1	.509	.204
TOTAL		1		.204

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0		
FK-ESII	WDF	8	.429	1.160
	WDG	8	.485	1.480
	WDH	2	.619	.602
	WDK	1	.769	.465
	WDN	8	5.552	.940
	WFB	2	.490	.378
	WLO	1	.627	.309
TOTAL		30		5.334
LX-ESII	WGB	4	.425	.568
	WGD	1	.484	.184
	WGG	1	.650	.342
	WGK	2	.789	.980
	WHB	2	.384	.232
TOTAL		10		2.306

Check Point 145

(looking North)

ATTACHMENT 9 SHEET 11



TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	2	.429	.290
	WDG	6	.485	1.110
	WDH	1	.619	.301
	WDI	1	.660	.342
	WLB	1	.509	.204
	WLC	1	.539	.229
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		20		4.230
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFD	1	.600	.283
	WGB	27	.425	3.854
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	3	.789	.147
	WGM	1	.874	.600
	WHB	5	.384	.580
	WHC	8	.405	1.032
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU	2	.376	.222
	WWN	2	.0172	.034
TOTAL		134		32.014

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	10	.353	.980
	WVA-1	22	.333	1.914
	WVB	9	.371	.972
TOTAL		41		3.866
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	29	.425	4.118
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHD	1	.439	.152
	WHE	2	.480	.362
	WHG	4	.519	.848
	WTO	8	.340	.728
	WTR	1	.360	.102
	BELDEN 8213	1	.405	.129
TOTAL		77		11.941
TL-ESIIL	WUB	50	.231	2.100
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		62		3.244

ATTACHMENT 9 SHEET 32

SAI-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TO AREA
SAI-ESII	WFB	8	.490	1.512					
	WFD	1	.600	.283					
	WGB	19	.425	2.698					
	WGC	4	.444	.620					
	WGD	2	.484	.368					
	WGE	2	.559	.492					
	WGI	2	.710	.792					
	WGK	3	.789	1.470					
	WHB	6	.384	.696					
	WHC	1	.405	.129					
	WHD	1	.439	.152					
	WHE	2	.480	.362					
	WHG	4	.519	.848					
	Belden 8213	1	.405	.129					
TOTAL		56		10.551					

25.013
TENTATIVE REVISION
OCTOBER 24, 1973

ATTACHMENT 13

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

POLYETHYLENE-INSULATED WIRE AND CABLE

A. SCOPE

*1. This specification covers single- and multiple-conductor wire and cable insulated with polyethylene, for installation in wet or dry locations with conductor temperatures not exceeding 75 C and for use in auxiliary power, control, and lighting circuits at voltages ranging from 600 to 8000 volts, 60 Hz AC, between conductors.

*2. All cable will be installed in conduits, in underground ducts, or on cable trays.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

*a. IPCEA-NEMA Standard S-61-402--NEMA WC5, IPCEA-NEMA Standards Publication, Thermoplastic-Insulated Wire and Cable. Specific references herein are from the Second Edition dated January 1968.

*b. IPCEA-NEMA Standard S-19-81--NEMA WC3, IPCEA-NEMA Standards Publication, Rubber-Insulated Wire and Cable. Specific references herein are from the Fifth Edition dated July 1969.

*c. All references herein are to IPCEA S-61-402 unless S-19-81 is mentioned specifically.

C. GENERAL REQUIREMENTS

*1. TVA type designation, voltage rating, conductor size, stranding if other than class B, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

2. Type designations are TVA designations and are described as follows:

*a. Type TW, single-conductor wire or cable, with polyethylene insulation and a nylon insulation jacket over the polyethylene.

C. GENERAL REQUIREMENTS, continued
C 2, continued

*b. Type PJ, single-conductor wire or cable, with polyethylene insulation and a polyvinyl-chloride insulation jacket over the polyethylene.

c. Type PMJ, multiple-conductor cable with a core of the specified number of type PM single conductors covered by a polyvinyl-chloride outer jacket.

d. Type PJJ, multiple-conductor cable with a core of the specified number of type PJ single conductors covered by a polyvinyl-chloride outer jacket.

*e. Type PSJ, single-conductor high-voltage cable with extruded strand and insulation shielding and with metallic electrostatic shielding.

*3. All outer covering of single-conductor wire or cable No. 8 AWG and larger and the outer jacket of all multiple-conductor cable shall have legible indent printing or legible permanent surface marking to show the following identification on approximately each foot of cable:

a. Manufacturer.

*b. The TVA contract number.

*c. TVA type designation (see section C.2. above) and voltage rating.

d. Size and number of conductors.

*4. Sequential footage marking on single-conductor cable, No. 8 AWG and larger, and on all multiconductor cable shall be supplied. The numerical markings shall be legible indent printing or legible permanent surface marking on the outside surface of the cable at 2-foot intervals and consist of 5-digit numbers increasing sequentially in increments of 2 feet (00020, 00022, 00024, etc.). The sequential markings may begin and end at any two numbers within the range of 00000 to 99998. Sequential footage markings shall be to a tolerance of ±2 percent.

*5. If indent printing is used for section C.3. or C.4. above, the depth of the indentations shall not be greater than 0.007 inch.

C. GENERAL REQUIREMENTS, continued

6. Each bidder shall submit with each copy of his bid complete information regarding the cable he proposes to furnish, including the following:

*a. Type designation and trade name of insulation, insulation jacket, and outer jacket materials.

b. Makeup of multiple-conductor cables.

*c. Performance records of representative installations of the same type and construction of wire and cable offered. If performance records have been submitted in a previous bid or are on file with TVA, an identifying reference shall be acceptable in lieu of a repetition of the data.

7. Color coding of multiple-conductor cable shall be furnished in accordance with IPCEA Section 5.6.3.1.1., Method 1, Table 5-1. All single-conductor wire or cable shall be black.

D. DETAILED REQUIREMENTS

1. Conductors shall be made of soft or annealed copper in conformity with IPCEA Part 2. All conductor sizes shall have concentric class B stranding and may be coated or uncoated.

*2. The insulation over each conductor shall consist of extruded heat-stabilized polyethylene, except that insulation not having a PVC jacket shall also be light-stabilized, applied concentrically about the conductor and shall meet the test requirements of section E.2.d. In-plant repairs of the insulation on shielded cable are prohibited unless specifically agreed to by TVA. All nominal polyethylene thickness shall conform to IPCEA Table 3-2, Column B or Ungrounded Neutral as applicable and except as noted in section D.5.

*3. The nylon insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of extruded nylon applied concentrically over the polyethylene. The nylon shall have a melting point substantially higher than that of polyethylene. Average thickness of the nylon jacket shall be not less than 4 mils and the minimum thickness shall not be less than 3 mils. No substitution of nylon for an equal thickness of polyethylene insulation shall be permitted.

*4. The polyvinyl chloride insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of a plasticized, noncontaminating, polyvinyl-chloride compound applied concentrically over the polyethylene. The insulation jacket compound shall meet the test

D. DETAILED REQUIREMENTS, continued
D 4, continued

requirements of section E.2.c. of this specification. The insulation jacket thickness shall meet the requirements of IPCEA Section 4.3.4. as applicable except as noted in section D.5.

*5. Single conductors of multiple-conductor cable with polyethylene insulation and polyvinyl chloride insulation jacket in sizes No. 14 through No. 9 AWG shall conform to IPCEA Section 7.4.3.2.2. The 20-mil polyethylene insulation and the 10-mil polyvinyl chloride insulation jacket shall have minimum thickness of not less than 18 mils and 8 mils respectively and the total thickness shall not be less than 27 mils.

*6. Shielding shall include conductor shielding, insulation shielding, and metallic tape shielding and shall be in conformance with IPCEA Sections 2.4. and 4.1., including test requirements, and the following specific requirements:

*a. Conductor shielding shall be extruded and shall consist of virgin black semiconducting thermoplastic material. The extruded shield shall have an average thickness of not less than 15 mils when measured over top of the strands, and a minimum thickness of not less than 12 mils. The outer surface of the conductor shield shall be cylindrical and shall be firmly bonded to the insulation. A semiconducting tape may be used between the conductor and the extruded semiconducting material if necessary to facilitate separation of shielding from conductor.

*b. Insulation shielding shall be extruded and shall consist of one layer of virgin, black, semiconducting thermoplastic material applied directly over the insulation. The average thickness of the insulation shield shall not be less than 30 mils.

*c. Metallic shielding tape shall be applied over the insulation shield. The metallic tape shall be lapped with minimum lap of 10 percent of tape width.

*7. Assembly of single conductors into multiple-conductor cables shall conform to IPCEA Section 5.2. or 5.3. and the following:

*a. Two-conductor cables may be assembled either parallel or helical

b. Cables with more than two conductors shall be assembled helical.

D. DETAILED REQUIREMENTS, continued
D 7, continued

*c. Thermoplastic nonhygroscopic fillers may be used on parallel assemblies and shall be used on helical assemblies to provide a substantially circular cross section. If necessary, a nonhygroscopic binding tape shall be provided to facilitate the removal of the outer jacket.

*8. An outer jacket of plasticized, noncontaminating, polyvinyl-chloride compound shall be applied concentrically and evenly over the core assembly of multiple-conductor cable or over the shield of shielded cable. The outside surface shall be reasonably smooth and free of irregularities. The jacket material shall meet the test requirements of section E.2.e. of this specification. The outer jacket thickness shall meet the requirements of IPCEA Section 4.3.4. as applicable. The normal calculated core diameter shall govern when manufacturing tolerances create a conflict about the proper diameter to follow. The color of the outer jacket shall be black.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping reels or coils from which samples shall be taken, the manufacturer shall take the necessary samples for measuring and testing at random from the shipping reels or coils of each lot of each item of cable he is offering for inspection.

E. INSPECTION AND TESTS, continued
E 2, continued

*b. Test samples for each of the following required tests shall be selected from at least 10 percent of the shipping lengths of each lot of each item offered for inspection. For nominal lengths less than 1800 feet, the number of samples shall be based on 10 percent of the equivalent lengths, as determined by dividing the total footage offered by 1800 feet; however, at least one sample per item shall be selected. Where a fraction results from the use of the 10 percent factor, a fraction less than 0.5 shall be dropped whereas a fraction 0.5 and greater shall require selecting an additional test sample.

*c. Dimensions shall be measured on 6-inch samples at a point free from any distortion or damage. For multiple-conductor cables, the samples shall be taken from the completely assembled cables and all the single conductors measured. Dimensions measured are to be shown in manufacturer's certified test reports and shall include diameter of conductor; thickness of insulation, thickness of extruded and metallic shielding, insulation jacket, and outer jacket; and overall diameter.

(1) Conductors shall be measured with a micrometer caliper reading to 0.001 inch, or equivalent. Acceptance or rejection shall be based on compliance with IPCEA Section 2.5.

*(2) Insulation, insulation jackets, metallic shielding, extruded semiconducting shielding, and outer jackets shall be measured with an Underwriters Laboratories, Incorporated, pin gauge, or equivalent, when fitted with a pin ranging in diameter from 0.025 to 0.020 inch. Acceptance or rejection shall be based on compliance with section D. of this specification.

(3) In case of doubt as to the accuracy of any measurement, the optical method shall be employed utilizing a micrometer microscope that is accurate to at least 0.0005 inch.

*d. Insulation shall be tested in accordance with IPCEA Section 3.9.1. except as follows:

*(1) Cold Bend Test at -35 C ±1 C in accordance with IPCEA Section 6.4.15. is required on all cables rated at 2001 volts and above.

E. INSPECTION AND TESTS, continued -
E 2 d, continued

*(2) For multiple-conductor cable samples, select one single conductor for testing when the sample contains 5 conductors or less and an additional single conductor for each 10 conductors or fraction thereof in excess of 5.

*e. Polyvinyl-chloride insulation jackets and outer jackets shall be tested in conformance with IPCEA Section 4.3.1. except as follows:

*(1) No tests other than Unaged Tensile and Elongation in accordance with IPCEA Sections 6.4.11.3. and 6.4.11.4. and (2) below are required on jackets less than 30 mils (0.76 mm) in thickness.

*(2) Physical - IPCEA Section 4.3.1., except the material shall meet the following minimum requirements:

<u>Physical Properties of Unaged Sample</u>

Minimum tensile strength, psi	- 1500
Minimum elongation at rupture, percent	- 250

<u>Physical Properties After Accelerated Aging</u>

After Air Oven Test at 119-121 C for	- 7 days
Tensile strength, minimum percent of unaged strength	- 90
Elongation, minimum percent of unaged sample	- 80

*(3) Flame tests - All completed multiple-conductor cables and single-conductor cables No. 8 AWG and larger shall pass the Vertical Flame Test, IPCEA S-19-81, Section 6.19.6. Both single conductors of multiple-conductor cables and single-conductor cables No. 9 AWG and smaller shall pass the Horizontal Flame Test, IPCEA S-19-81, Section 6.13.2. No flame test is required on nylon-jacketed single-conductor cable or on nylon-jacketed single-conductors of multiple-conductor cables.

*(4) For multiple-conductor cable, the outer jacket and the jacket of only one conductor from the selected sample shall be teste

E. INSPECTION AND TESTS, continued

*3. Factory Tests on Shipping Reel and Coil Lengths

*a. Alternating-current high-voltage tests shall be made on each length of each lot of each item in conformance with IPCEA Section 6.11. and the following requirements:

*(1) A test in water of all completed nonshielded, single-conductor cable shall be made in accordance with IPCEA Sections 6.11.3. and 6.11.4. and Table 3-2, except that for cable rated at 600 volts the minimum AC test voltage shall not be less than 5000 volts for any size conductor.

*(2) On multiple-conductor cable a preliminary test of each single conductor shall be made in accordance with paragraph (1) above. After assembly and final covering, each conductor shall be tested in accordance with IPCEA Section 6.11.4.

*(3) Shielded cable shall be tested in accordance with IPCEA Sections 6.11.3. and 6.11.5. The test voltage shall be applied between conductor and shield and shall be that specified in IPCEA Table 3-2 for cable with ungrounded neutral.

*(4) Shielded cable shall be tested for corona level in accordance with IPCEA Sections 3.9.2.4. and 6.13. The minimum corona extinction level shall be that required for cable with ungrounded neutral.

*b. Insulation resistance tests shall be made on each shipping reel or coil length in conformity with IPCEA Section 6.12. Any cable length which fails to meet the specified insulation resistance shall be rejected.

*c. Direct-current high-voltage tests shall be made on each shipping length of each lot of each item rated at 2001 volts and higher in conformance with IPCEA Sections 3.9.2.3. and 6.11.2. The direct-current test voltage shall be applied between the conductor and the shield. The DC test voltage shall be 3.0 times the AC test voltage as required by IPCEA Table 3-2 for cable with ungrounded neutral.

E. INSPECTION AND TESTS, continued
E 3, continued

*d. Conductor resistance tests shall be made on 10 percent of the shipping lengths of each lot of each item ready at the time of inspection and the results expressed on ohms per 1000 feet at either 20 or 25 C. If the resistance of any length tested exceeds the specified resistance allowed by IPCEA Section 2.6., that length may be retested. If the resistance is still found to be in excess, the length shall be rejected; each length of that lot shall be tested, and any length found to be in excess shall also be rejected.

*e. Shield resistance shall be measured on each shipping length of each lot of each item of shielded cable and expressed in ohms per 1000 feet.

4. Factory Test Reports

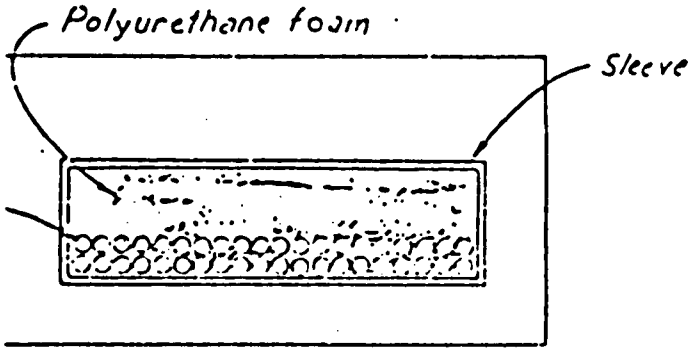
*a. The Contractor shall furnish seven certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number(s), mark number(s), shipping reel(s) number(s), date of test, TVA cable type and voltage rating, and the number and size of conductors. The report shall include the results of all the tests and measurements specified herein.

F. PACKING AND MARKING

*1. The material shall be packed, sealed, and shipped in accordance with ANSI MH5.2.2-1972, Level D classification. Wire or cable shall be shipped on reels or in coils in continuous lengths as called for in the Schedule of Prices. The reels, if wooden, shall be painted or treated in such a way as to resist damage from prolonged exposure to an outdoor environment.

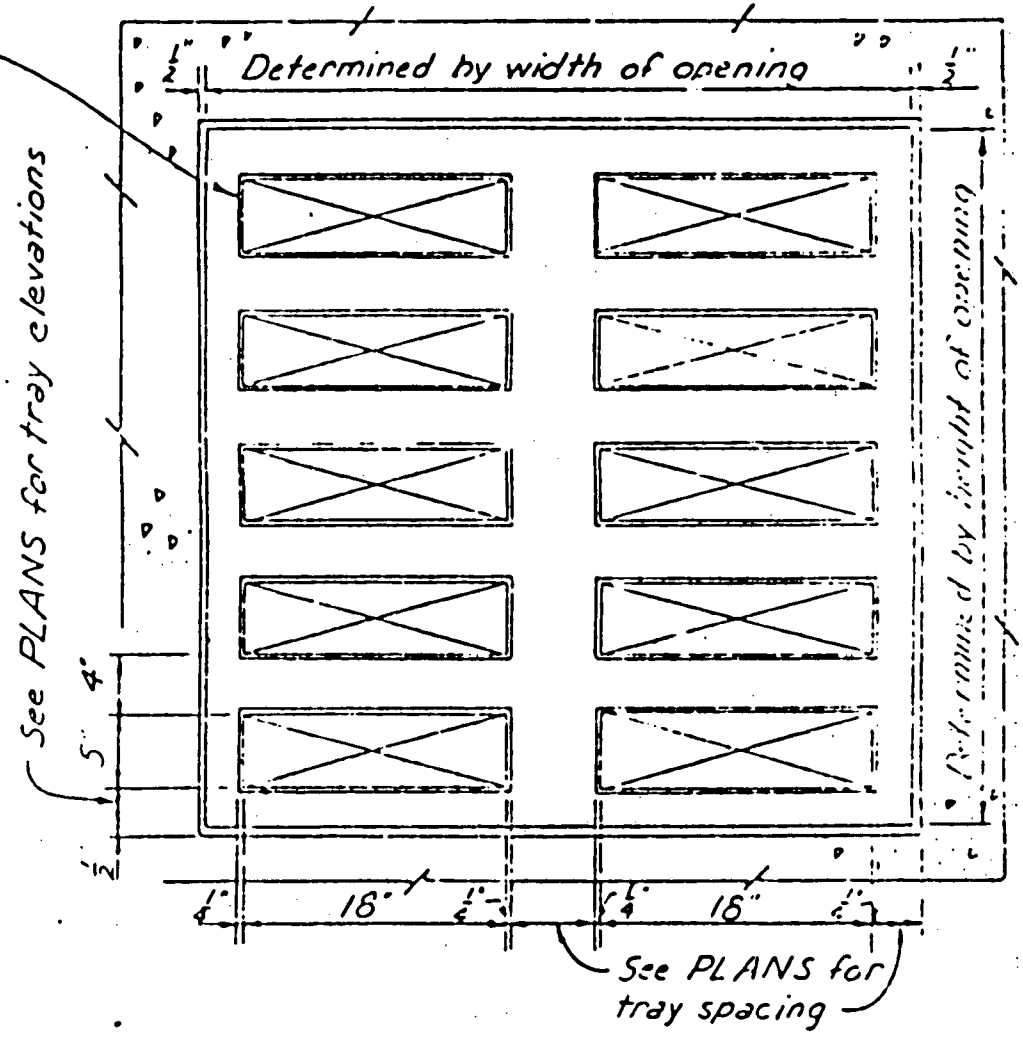
*2. Reels and coils shall be plainly marked with the contract number, item number, reel number, length shipped on reel or coil, number and size of conductors, TVA insulation designation and voltage rating, year of manufacture, and name of manufacturer. All labels and tags shall be waterproofed and fastened securely to the reel or coil.

*Revisions since 9-15-70.

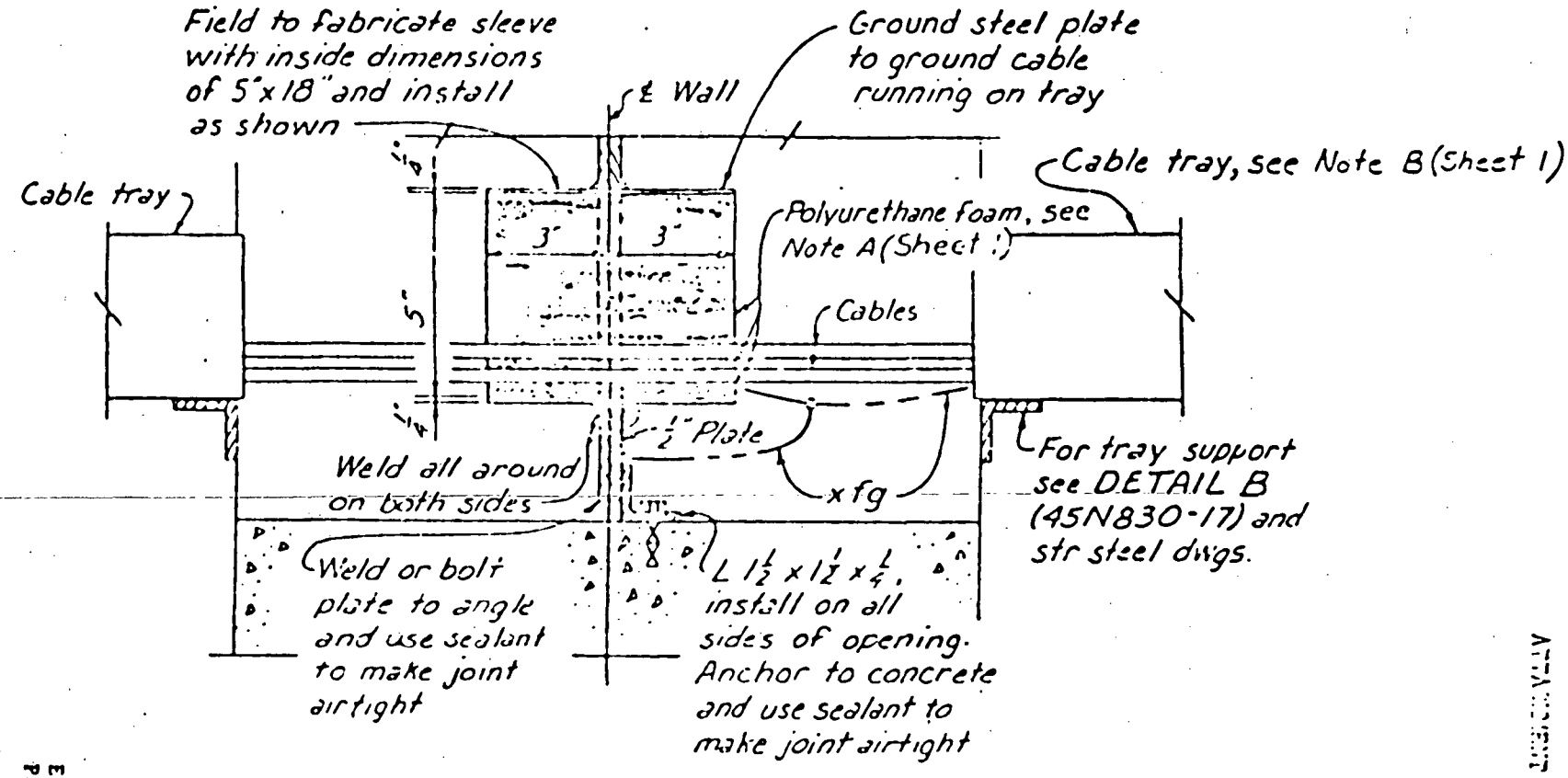


NOTES:

1. When all cables have been installed through cable sleeve, seal the remaining opening and voids with 6" ± of Urethane foam or equal before applying flame proofing compound, Flamastic 71A or equal.
2. Apply approximately 1/8" to 1/4" of flame proofing compound on the steel sleeve and on both top and bottom of the tray and cables for 12" on both sides of the barrier.



FRONT ELEVATION
Scale: 1" = 1'-0"



SIDE VIEW
Scale: 3"=1'-0"

ATTACHMENT 11

21.000
TENTATIVE REVISION
JANUARY 24, 1975

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

RIGID ALUMINUM CONDUIT

A. SCOPE

*1. This specification covers rigid aluminum conduit and accessories such as couplings, elbows, and fittings for use as metal raceways for wires and cables.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

Underwriters Laboratories, Incorporated,
Standard UL6 - Rigid Metallic Conduit

ANSI C80.5 - Rigid Aluminum Conduit

*ANSI B2.1 - Standard Pipe Threads

C. GENERAL REQUIREMENTS

*1. Size, quantity, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

D. DETAILED REQUIREMENTS

*1. The conduit and accessory items shall be manufactured from 6063 aluminum alloy, T-1 temper with maximum copper content not exceeding four-tenths of 1 percent. Conduit shall be furnished in nominal 10-foot lengths. Each length shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1. One end of each length of conduit shall be fitted with a coupling and the end opposite the coupling shall be fitted with a plastic color-coded thread protector.

*2. Elbows shall be furnished in 6063 aluminum alloy, T-1 temper, and shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1.

*Revisions since 7-18-62.

D. DETAILED REQUIREMENTS, continued

*3. Couplings shall be furnished in 6063 aluminum alloy, T-1 temper, and shall be internally threaded with American National Standards Institute straight pipe thread, ANSI B2.1.

*4. Color-coded thread protectors furnished with aluminum conduit shall be color-coded as follows:

Red - Quarter-inch sizes
Black - Half-inch sizes
Blue - Inch sizes

*5. Specification Compliance

*a. Conduit, elbows, and couplings furnished under this specification shall conform to the requirements of Underwriters Laboratories, Incorporated, Standard UL6, and American National Standards Institute C80.5.

*b. In addition to the manufacturer's name or trademark, as required by Underwriters Laboratories, Incorporated, Standard UL6, each length of conduit and elbow shall have the Underwriters Laboratories, Incorporated, label or an acceptable equivalent, firmly attached.

*6. Conformance labels shall be attached to the conduit with pressure-sensitive adhesive or similar means to provide easy removal without scarring the conduit, or shall be attached to the thread protectors. Stenciling or permanent imprint in the metal, with natural finish, will be accepted.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*Revisions since 7-18-62.

E. INSPECTION AND TESTS, continued
E 1, continued

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

*c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping bundles or packages from which samples shall be taken, the manufacturer shall take the necessary sample(s) for measuring and testing at random from the shipping bundles or packages of each item of conduit, elbow, or coupling. At least one sample per item shall be taken.

*b. Conduit shall be tested for ductility at a temperature of 72 F ±4 F in accordance with ANSI C80.5, Part 5.

*c. Dimensions of conduit (inside diameter, outside diameter, and wall thickness) shall be measured with a micrometer caliper reading 0.001 inch or equivalent and shall conform to ANSI C80.5, Table 2.

*d. Conduit shall be measured for length, without coupling, and for weight of ten units with one coupling attached to each length and shall conform to ANSI C80.5, Table 2.

*e. Couplings shall be measured for outside diameter and minimum length and shall conform to ANSI C80.5, Table 3.

*f. Elbows shall be measured for radius to center of conduit and length of straight section at each end and shall conform to ANSI C80.5, Table 4.

3. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number, mark number, and date of test. The report shall include the results of all the tests and measurements specified herein.

*Revisions since 7-18-62.

E. INSPECTION AND TESTS, continued

4. TVA, at its discretion, may waive any or all of the provisions for inspection, tests, and test reports, either in the invitation for bids, at time of award of contract, or at any time thereafter.

F. PACKING AND MARKING

1. The conduit and accessories shall be packaged, or bundled, and wrapped to prevent damage and marked in accordance with standard commercial practice so as to ensure acceptance by common carrier for safe transportation at the lowest rate to the point of delivery.

*2. Shipping bundles or packages shall be plainly marked with the contract number, item number, name of material, number and size of items contained therein, and the name of the Contractor. All labels and tags shall be waterproofed and fastened securely to the bundles or packages.

*Revisions since 7-18-62.

ATTACHMENT 12

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

RIGID STEEL CONDUIT (ZINC COATED)

21.0001
TENTATIVE REVISION
MAY 23, 1973

A. SCOPE

*1. This specification covers rigid steel conduit (zinc coated) and accessories such as couplings, elbows, and fittings for use as metal raceways for wires and cables.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

Underwriters Laboratories, Incorporated, Standard - UL6 - Rigid Metal Conduit

ANSI C80.1 - Rigid Steel Conduit (Zinc Coated)
*ANSI B2.1 - Standard Pipe Threads

C. GENERAL REQUIREMENTS

*1. Size, quantity, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specificat! are stated in the Schedule of Prices.

D. DETAILED REQUIREMENTS

*1. The conduit and accessory items shall be manufactured from mild steel tube with a zinc coating having a minimum thickness of 0.0008 inch on both inside and outside surfaces. Conduit shall be furnished in nominal 10-foot lengths. Each length shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1., and shall have a uniform coating of zinc applied to the threads. One end of each length of conduit shall be fitted with a coupling and the end opposite the coupling shall be fitted with a plastic color-coded thread protector.

*2. Elbows shall be furnished in mild steel tube with a zinc coating on both inside and outside surfaces and shall be threaded on both ends with American National Standards Institute tapered pipe thread, ANSI B2.1. A UL listed rust preventive coating shall be applied to the threads.

*Revisions since 5-7-76.

D. DETAILED REQUIREMENTS, continued

*3. Couplings shall be furnished in mild steel tube with zinc coating and shall be internally threaded with American National Standards Institute straight-pipe thread, ANSI B2.1. A uniform zinc coating shall be applied to the internal threads.

*4. Color-coded thread protectors furnished with rigid steel conduit shall be color coded as follows:

Red - Quarter-inch sizes
Black - Half-inch sizes
Blue - Inch sizes

*5. Specification Compliance

*a. Conduit, elbows, and couplings furnished under this specification shall conform to the requirements of Underwriters Laboratories, Incorporated, Standard UL6, and American National Standards Institute C80.1.

*b. In addition to the manufacturer's name or trademark, as required by Underwriters Laboratories, Incorporated, Standard UL6, each length of conduit and elbow shall have the Underwriters Laboratories, Incorporated, label or an acceptable equivalent, firmly attached.

*6. Conformance labels shall be attached to the conduit with pressure-sensitive adhesive or similar means to provide easy removal without scarring the conduit, or shall be attached to the thread protectors. Stenciling or permanent imprint in the metal, with natural finish, will be accepted.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

E. INSPECTION AND TESTS, continued
E 1, continued

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

*c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to be that a TVA representative will designate the shipping bundles or packages from which samples shall be taken, the manufacturer shall take the necessary samples(s) for measuring and testing at random from the shipping bundles or packages of each item of conduit, elbow, or coupling. At least one sample per item shall be taken.

*b. Conduit shall be tested for ductility at a temperature of 72 F ±4 F in accordance with ANSI C80.1.

*c. Conduit and accessory items shall be tested for thickness of zinc coating in accordance with ANSI C80.1, Paragraph 5.2, Method 3 (Preece test). The material shall withstand four 1-minute immersions.

*d. Dimensions of conduit (inside diameter, outside diameter, and wall thickness) shall be measured with a micrometer caliper reading 0.001 inch or equivalent and shall conform to ANSI C80.1, Table 2.

*e. Conduit shall be measured for length, without coupling, and for weight of 10 units with one coupling attached to each length and shall conform to ANSI C80.1, Table 2.

*f. Couplings shall be measured for outside diameter and minimum length and shall conform to ANSI C80.1, Table 3.

*g. Elbows shall be measured for radius to center of conduit and length of straight section at each end and shall conform to ANSI C80.1, Table 4.

*Revisions since 6-7-66.

E. INSPECTION AND TESTS, continued

3. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number, mark number, and date of test. The report shall include the results of all the tests and measurements specified herein.

*4. TVA, at its discretion, may waive any or all of the provisions for inspection, tests, and test reports, either in the invitation for bids, at time of award of contract or at any time thereafter.

F. PACKING AND MARKING

1. The conduit and accessories shall be packaged, or bundled, and wrapped to prevent damage and marked in accordance with standard commercial practice so as to ensure acceptance by common carrier for safe transportation at the lowest rate to the point of delivery.

*2. Shipping bundles or packages shall be plainly marked with the contract number, item number, name of material, number and size of items contained therein, and the name of the Contractor. All labels and tags shall be waterproofed and fastened securely to the bundles or packages.

ATTACHMENT 14

25.016
TENTATIVE REVISION
MARCH 16, 1973

TENNESSEE VALLEY AUTHORITY

STANDARD SPECIFICATION

FOR

CROSS-LINKED POLYETHYLENE-INSULATED WIRE AND CABLE

A. SCOPE

*1. This specification covers single- and multiple-conductor wire and cable insulated with cross-linked polyethylene, for installation in wet or dry locations with conductor temperatures not exceeding 90 C and for use in auxiliary power, control, and lighting circuits at voltages ranging from 600 to 15,000 volts, 60 Hz AC, between conductors.

*2. All cable will be installed in conduits, in underground ducts, or on cable trays.

B. APPLICABLE SPECIFICATIONS

*1. The current revision to the following specifications of the issue in effect on date of the invitation to bid shall form a part of this specification:

*a. IPCEA-NEMA Standard S-66-524--NEMA WC7, IPCEA-NEMA Standards Publication, Cross-Linked-Thermosetting-Polyethylene-Insulated Wire and Cable. Specific references herein are from the First Edition dated May 1971.

*b. IPCEA-NEMA Standard S-19-81--NEMA WC3, IPCEA-NEMA Standards Publication, Rubber-Insulated Wire and Cable. Specific references herein are from the Fifth Edition dated July 1969.

*c. All references herein are to IPCEA S-66-524 unless S-19-81 is mentioned specifically.

C. GENERAL REQUIREMENTS

*1. TVA type designation, voltage rating, conductor size, stranding if other than class B, special Quality Assurance Procedures if required, and any other specific requirements or exceptions to this specification are stated in the Schedule of Prices.

2. Type designations are TVA designations and are described as follows:

*a. Type CP, single-conductor wire or cable, with cross-linked polyethylene insulation.

C. GENERAL REQUIREMENTS, continued
C 2, continued

*b. Type CPJ, single-conductor wire or cable, with cross-linked polyethylene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethylene.

c. Type CPJJ, multiple-conductor cable with a core of the specified number of type CPJ single conductors covered by a polyvinyl-chloride outer jacket.

d. Type CPMJ, multiple-conductor cable with a core of the specified number of type CP single conductors covered by a polyvinyl-chloride outer jacket.

*e. Type CPSJ, single-conductor high-voltage cable with extruded strand and insulation shielding and with metallic electrostatic shielding.

*3. All outer covering of single-conductor wire or cable No. 8 AWG and larger and the outer jacket of all multiple-conductor cable shall have legible indent printing or legible permanent surface marking to show the following identification on approximately each foot of cable:

a. Manufacturer.

b. Year of manufacture.

*c. TVA type designation (see section C.2. above) and voltage rating.

d. Size and number of conductors.

*4. Sequential footage marking on single-conductor cable, No. 8 AWG and larger, and on all multiconductor cable shall be supplied. The numerical markings shall be legible indent printing or legible permanent surface marking on the outside surface of the cable at 2-foot intervals and consist of 5-digit numbers increasing sequentially in increments of 2 feet (00020, 00022, 00024, etc.). The sequential markings may begin and end at any two numbers within the range of 00000 to 99993. Sequential footage markings shall be to a tolerance of ±2 percent.

*5. If indent printing is used for section C.3. or C.4. above, the depth of the indentations shall not be greater than 0.007 inch.

*6. At the option of the manufacturer, on 15-kv class cable a marker tape under the outer jacket containing information required in sections C.3. and C.4. may be substituted.

C. GENERAL REQUIREMENTS, continued

7. Each bidder shall submit with each copy of his bid complete information regarding the cable he proposes to furnish, including the following:

*a. Type designation and trade name of insulation, insulation jacket, and outer jacket materials.

b. Makeup of multiple-conductor cables.

*c. Performance records of representative installations of the same type and construction of wire and cable offered. If performance records have been submitted in a previous bid or are on file with TVA, an identifying reference shall be acceptable in lieu of a repetition of the data.

*8. Color coding of multiple-conductor cable shall be furnished in accordance with IPCEA S-19-81, Section 5.6.3.1.1., Method 1, Table 5-2. All single-conductor wire or cable shall be black.

D. DETAILED REQUIREMENTS

1. Conductors shall be made of soft or annealed copper in conformity with IPCEA Part 2. All conductor sizes shall have concentric class B stranding and may be coated or uncoated.

*2. The insulation over each conductor shall consist of extruded heat-stabilized cross-linked polyethylene, except that insulation not having a jacket shall also be light-stabilized, applied concentrically about the conductor and shall meet the test requirements of section E.2.d. In-plant repairs of the insulation on shielded cable are prohibited unless specifically agreed to by TVA. All nominal cross-linked polyethylene thickness shall conform to IPCEA Table 3-1 as noted below:

*a. Type CP single-conductor cable and individual conductors of type CPJJ multiple-conductor cable shall conform to Column A.

*b. Type CPJ single-conductor cable and individual conductors of type CPJJ multiple-conductor cable shall conform to Column B.

*c. Type CPSJ single-conductor shielded cable shall conform to the column marked Ungrounded Neutral (Shielded).

*3. The insulation jacket covering the insulation on single-conductor cable or single conductors of multiple-conductor cable shall consist of a plasticized, noncontaminating, polyvinyl-chloride compound

D. DETAILED REQUIREMENTS, continued
D 3, continued

applied concentrically over the cross-linked polyethylene. The insulation jacket compound shall meet the test requirements of section E.2.e. of this specification. The insulation jacket thickness shall meet the requirements of IPCEA Section 4.3.3. as applicable.

*4. Shielding shall include conductor shielding, insulation shielding, and metallic tape shielding and shall be in conformance with IPCEA Sections 2.4. and 4.1., including test requirements, and the following specific requirements:

*a. Conductor shielding shall be extruded and shall consist of virgin black semiconducting thermosetting material. The extruded shield shall have an average thickness of not less than 15 mils when measured over top of the strands, and a minimum thickness of not less than 12 mils. The outer surface of the conductor shield shall be cylindrical and shall be firmly bonded to the insulation. A semiconducting tape may be used between the conductor and the extruded semiconducting material if necessary to facilitate separation of shielding from conductor.

*b. Insulation shielding shall be extruded and shall consist of one layer of virgin, black, semiconducting thermosetting material applied directly over the insulation. The average thickness of the insulation shield shall not be less than 30 mils.

*c. Metallic shielding tape shall be applied over the insulation shield. The metallic tape shall be lapped with minimum lap of 10 percent of tape width.

*5. Assembly of single conductors into multiple-conductor cables shall conform to IPCEA Section 5.2. or 5.3. and the following:

*a. Two-conductor cables may be assembled either parallel or helical.

b. Cables with more than two conductors shall be assembled helical

*c. Thermoplastic nonhygroscopic fillers may be used on parallel assemblies and shall be used on helical assemblies to provide a substantially circular cross section. If necessary, a nonhygroscopic binding tape shall be provided to facilitate the removal of the outer jacket.

D. DETAILED REQUIREMENTS, continued

*6. An outer jacket of plasticized, noncontaminating, polyvinyl-chloride compound shall be applied concentrically and evenly over the core assembly of multiple-conductor cable or over the shield of shielded cable. The outside surface shall be reasonably smooth and free of irregularities. The jacket material shall meet the test requirements of section E.2.e. of this specification. The outer jacket thickness shall meet the requirements of IPCEA Section 4.3.3. as applicable. The normal calculated core diameter shall govern when manufacturing tolerances create a conflict about the proper diameter to follow. The color of the outer jacket shall be black.

E. INSPECTION AND TESTS

1. Factory Inspection

*a. No material covered by this specification shall be shipped from its point of manufacture before it has been inspected by a TVA Inspector, unless TVA authorizes inspection to be made elsewhere or waived. The TVA Inspector shall have the necessary access and facilities at the Contractor's factory for unrestricted inspection of the work. A schedule of production of TVA's material shall be furnished to the TVA Inspector on his request.

*b. Whether witnessed or not by TVA, the manufacturer shall make the inspection and tests specified herein and shall submit the results in his certified test reports.

c. The release of any material by a TVA Inspector shall in no way relieve the Contractor of any of his responsibility for meeting all of the requirements of this specification, and shall not prevent subsequent rejection if such material is later found to be defective.

2. Factory Tests of Sample Lengths

*a. Unless notified by TVA in writing before factory tests are to begin that a TVA representative will designate the shipping reels or coils from which samples shall be taken, the manufacturer shall take the necessary samples for measuring and testing at random from the shipping reels or coils of each lot of each item of cable he is offering for inspection.

*b. Test samples for each of the following required tests shall be selected from at least 10 percent of the shipping lengths of each lot of each item offered for inspection. For nominal lengths less than 1800 feet, the number of samples shall be based on 10 percent of the equivalent lengths, as determined by dividing the total footage offered by 1800 feet; however, at

*Revisions since 12-10-70

E. INSPECTION AND TESTS, continued
E 2 b, continued

least one sample per item shall be selected. Where a fraction results from the use of the 10 percent factor, a fraction less than 0.5 shall be dropped whereas a fraction 0.5 and greater shall require selecting an additional test sample.

*c. Dimensions shall be measured on 6-inch samples at a point free from any distortion or damage. For multiple-conductor cables, the samples shall be taken from the completely assembled cables and all the single conductors measured. Dimensions measured are to be shown in manufacturer's certified test reports and shall include diameter of conductor; thickness of insulation, thickness of extruded and metallic shielding, insulation jacket, and outer jacket; and overall diameter.

(1) Conductors shall be measured with a micrometer caliper reading to 0.001 inch, or equivalent. Acceptance or rejection shall be based on compliance with IPCEA Section 2.5.

*(2) Insulation, insulation jackets, metallic shielding, extruded semiconducting shielding, and outer jackets shall be measured with an Underwriters Laboratories, Incorporated, pin gauge, or equivalent, when fitted with a pin ranging in diameter from 0.025 to 0.020 inch. Acceptance or rejection shall be based on compliance with section D. of this specification.

(3) In case of doubt as to the accuracy of any measurement, the optical method shall be employed utilizing a micrometer microscope that is accurate to at least 0.0005 inch.

*d. Insulation shall be tested in accordance with IPCEA Sections 3.6. and 3.7.1. except as follows:

*(1) Cold Bend Test at -35 C ±1 C in accordance with IPCEA Section 6.10.3 is required on all cables rated at 2001 volts and above.

*(2) For multiple-conductor cable samples, select one single conductor for testing when the sample contains 5 conductors or less and an additional single conductor for each 10 conductors or fraction thereof in excess of 5.

E. INSPECTION AND TESTS, continued
E 2, continued

*e. Polyvinyl-chloride insulation jackets and outer jackets shall be tested in conformance with IPCEA Section 4.3.1. except as follows:

*(1) No tests other than Unaged Tensile and Elongation in accordance with IPCEA Sections 6.4.11.3. and 6.4.11.5. and (2) below are required on jackets less than 30 mils (0.76 mm) in thickness.

*(2) Physical - IPCEA Section 4.3.1., except the material shall meet the following minimum requirements:

Physical Properties of Unaged Sample

Minimum tensile strength, psi	- 1500
Minimum elongation at rupture, percent	- 250

Physical Properties After Accelerated Aging

After Air Oven Test at 119-121 C for	- 7 days
Tensile strength, minimum percent of unaged strength	- 90
Elongation, minimum percent of unaged sample	- 80

*(3) Flame tests - All completed multiple-conductor cables and single-conductor cables No. 8 AWG and larger shall pass the Vertical Flame Test, IPCEA S-19-81, Section 6.19.6. Both single conductors of multiple-conductor cables and single-conductor cables No. 9 AWG and smaller shall pass the Horizontal Flame Test, IPCEA S-19-81, Section 6.13.2.

*(4) For multiple-conductor cable, the outer jacket and the jacket of only one conductor from the selected sample shall be tested

*3. Factory Tests on Shipping Reel and Coil Lengths

*a. Alternating-current high-voltage tests shall be made on each length of each lot of each item in conformance with IPCEA Section 6.14. and the following requirements:

*(1) A test in water of all completed nonshielded, single-conductor cable shall be made in accordance with IPCEA Sections 6.14.3. and 6.14.4. and Table 3-1, except that for cable rated at 600 volts the minimum AC test voltage shall not be less than 5000 volts for any size conductor.

E. INSPECTION AND TESTS, continued
E 3 a, continued

*(2) On multiple-conductor cable a preliminary test of each single conductor shall be made in accordance with paragraph (1) above. After assembly and final covering, each conductor shall be tested in accordance with IPCEA Section 6.14.4.

*(3) Shielded cable shall be tested in accordance with IPCEA Sections 6.14.3. and 6.14.5. The test voltage shall be applied between conductor and shield and shall be that specified in IPCEA Table 3-1 for cable with ungrounded neutral.

*(4) Shielded cable shall be tested for corona level in accordance with IPCEA Sections 3.7.2.4. and 6.16. The minimum corona extinction level shall be that required for cable with ungrounded neutral.

*b. Insulation resistance tests shall be made on each shipping reel or coil length in conformity with IPCEA Section 6.15. Any cable length which fails to meet the specified insulation resistance shall be rejected.

*c. Direct-current high-voltage tests shall be made on each shipping length of each lot of each item rated at 2001 volts and higher in conformance with IPCEA Sections 3.7.2.3. and 6.14.2. The direct-current test voltage shall be applied between the conductor and the shield. The DC test voltage shall be 3.0 times the AC test voltage as required by IPCEA Table 3-1 for cable with ungrounded neutral.

*d. Conductor resistance tests shall be made on 10 percent of the shipping lengths of each lot of each item ready at the time of inspection and the results expressed in ohms per 1000 feet at either 20 or 25 C. If the resistance of any length tested exceeds the specified resistance allowed by IPCEA Section 2.6., that length may be retested. If the resistance is still found to be in excess, the length shall be rejected; each length of that lot shall be tested, and any length found to be in excess shall also be rejected.

*e. Shield resistance shall be measured on each shipping length of each lot of each item of shielded cable and expressed in ohms per 1000 feet.

E. INSPECTION AND TESTS, continued

4. Factory Test Reports

*a. The Contractor shall furnish five certified copies of each factory test report to TVA promptly after conclusion of tests. Each test report shall be identified by TVA contract number, item number(s), mark number(s), shipping reel(s) number(s), date of test, TVA cable type and voltage rating, and the number and size of conductors. The report shall include the results of all the tests and measurements specified herein.

F. PACKING AND MARKING

1. The material shall be packed, sealed, and shipped in accordance with recognized standard practice. Wire or cable shall be shipped on reels or in coils in continuous lengths as called for in the Schedule of Prices.

*2. Reels and coils shall be plainly marked with the contract number, item number, reel number, length shipped on reel or coil, number and size of conductors, TVA insulation designation and voltage rating, year of manufacture, and Name of Manufacturer. All labels and tags shall be waterproofed and fastened securely to the reel or coil.

Table I

4 KV Shutdown Boards and Loads

4 KV Shutdown Board A Loads

1. Core Spray (CS) Pump Motors 1A and 2A
2. Residual Heat Removal (RHR) Pump Motors 1A and 2A
3. RHR Service Water Pumps Motors A2 and A3
4. Raw Cooling Water Pump 1D
5. Control Rod Drive (CRD) Pump 1B
6. 480 Volt Shutdown Board 1A Transformer (Normal)
7. 480 Volt Diesel Auxiliary Board A and B Transformer
8. Fire Pump A

4 KV Shutdown Board B Loads

1. CS Pump Motors 1C and 2C
2. RHR Pump Motors 1C and 2C
3. RHR Service Water Pumps Motors C3 and C2
4. 480 Volt Shutdown Board 1A and 1B Transformer (Backup)
5. 480 Volt Diesel Auxiliary Board B Transformer
6. 480 Volt Shutdown Board 2A Transformer (Normal)
7. Fire Pump B

4 KV Shutdown Board C Loads

1. CS Pump Motors 1B and 2B
2. RHR Pump Motors 1B and 2B
3. RHR Service Water Pump Motors B1 and B2
4. 480 Volt Shutdown Board 2A and 2B Transformer (Backup)
5. 480 Volt Shutdown Board 1B Transformer (Normal)
6. Fire Pump C

4 KV Shutdown Board D Loads

1. CS Pump Motors 1D and 2D
2. RHR Pump Motors 1D and 2D
3. RHR Service Water Pump Motors D1 and D2
4. 480 Volt Diesel Auxiliary Board A Transformer
5. 480 Volt Shutdown Board 2B Transformer (Normal)

Table II

480 Volt Shutdown Boards Loads

480 Volt Shutdown Board 1A Loads

1. Unit Preferred M-G Set 1
2. Instrumentation and Control (I&C) Bus A Transfo
3. Control Bay Vent Board A (Emergency)
4. 480 Volt Motor Operated Valve (MOV) Board 1A (Normal)
5. 480 Volt MOV Board 1B (Emergency)
6. 480 Volt MOV Board 1C (Auto throwover) (Emergency)
7. 250 Volt Battery Charger
8. Control and Service Air Compresser 1A
9. Control Bay Water Chiller A
10. Closed Cooling Water (CCW) Pump 1A
11. Reactor Water Cleanup (RWCU) Recirculating Pump 1A
12. Recirculating Pumps M-G Set Oil Pumps
13. Standby Liquid Control (SLC) Pump 1A
14. Fuel Pool Co[illegible] (FPC) Pump 1A
15. Drywell Blowers 1A-1 and 1B-1

480 Volt Shutdown Board 1B Loads

1. I&C Bus B Transformer
2. Unit Preferred Transformer TUP 1
3. 480 Volt Condensate Demineralizer Board (Emergency)
4. 480 Volt ReactorMOV Board 1A (Emergency)
5. 480 Volt Reactor MOV Boards 1B and 1C (Normal)
6. Turning Gear Oil Pump

7. Control Bay Water Chiller B
8. CCW Pumps 1B and 1C
9. RWCU Recirculating Pump 1B
10. Recirculating Pump M-G Set Oil Pumps
11. SLC Pump 1B
12. FPC Pump 1B
13. Drywell Blowers 1A-2 and 1B-2

480 Volt Shutdown Board 2A Loads

1. Unit Preferred M-G set 2
2. I&C Bus 2A Transformer
3. 480 Volt Diesel Auxiliary Board 3EA (Normal
4. Control Bay Vent Board B (Emergency)
5. 480 Volt Reactor MOV Board 2A (Normal)
6. 480 Volt Reactor MOV Boards 2B and Board 2C (Emergency)
7. 250 Volt Battery Charger 2A
8. Control and Service Air Compressor D
9. CCW Pump 2A
10. RWCU Pump 2A
11. Recirculating Pump M-G Set Oil Pumps
12. SLC Pump 2A
13. FPC Pump 2A
14. Drywell Blowers 2A-1 and 2B-1

480 Volt Shutdown Board 2B Loads

1. I&C Bus B Transformer
2. Unit Preferred Transformer
3. 480 Volt Condensate Demineralizer Board 2 (Emergency)

4. 480 Volt Reactor MOV Board 2A (Emergency)
5. 480 Volt Reactor MOV Board 2B and 2C (Normal)
6. 250 Volt Battery Charter 2B
7. Turning Gear Oil Pump
8. Raw Cooling Water Booster Pump 2A
9. CCW Pump 2B
10. RWCU Recirculating Pump 2B
11. Recirculating Pump M-G set oil pumps 2A-2 and 2B-2
12. SLC Pump 2B
13. FPC Pump 2B
14. Drywell Blowers 2A-2 and 2B-2

Table III

Reactor MOV Board Loads

480 Volt Reactor MOV Board 1A

1. Turbine Building Emergency Lighting Transformer
2. Reactor Building Emergency Lighting Transformer
3. "A" and "C" CS Systems Valves
4. "A" and "C" RHR Systems Valves
5. Drywell Control Air System 1A
6. Reactor Protection System (RPS) M-G set 1A
7. RWCU Isolation Valve FCV 69-1
8. High Pressure Coolant Injection (HPCI) Steam Isolation Valve 73-2
9. SLC Trace Heating (Normal)
10. Drywell Equipment and Floor Drain Sump Pumps 1A
11. Drywell Blowers 1A-3 and 1B-3
12. Turbine Turning Gear Motor
13. Exhaust Fans 1A Shutdown Board Room

480 Volt Reactor MOV Board 2A

1. Loads for Unit 2 that are comparable to those above.

480 Volt Reactor MOV Board 1B

1. Annunciator Battery Chargers
2. Reactor Building Emergency Lighting Transformer
3. RPS M-G Set 1B
4. CS System Valves B and D
5. RHR System Valves B and D
6. Drywell Control Air System 1B

7. CCW Sectionalizing Valve FC70-48

8. CCW Isolation Valve FC70-47

9. RWCU Isolation Valve FC69-12

10. Reactor Core Isolation Cooling (RCIC) System Steam Isolation Valve FCV71-2

11. Units 1 and 2 Standby Coolant Valve

12. RWCU Holding Pump 1B

13. Drywell Equipment and Floor Drain Sump Pumps 1B

14. Drywell Blowers 1A-4 and 1B-4

15. Turbine Bearing Lift Pumps

16. SLC Trace Heating (alternate supply)

17. All cross tie valves in RHR System

18. Various fans and sump pumps

480 Volt Reactor MOV Board 2B

1. Loads for Unit 2 that are comparable to those above.

480 Volt Reactor MOV Board 1C

1. CRD System Valves

2. RWCU System Valves

3. Drywell Blowers 1A-5 and 1B-5

4. Miscellaneous drain valves

5. Miscellaneous sump pumps

6. Cooling water valves for turbine equipment

7. Recirculating System Valves

8. RHR Inboard Valve RCV75-53

480 Volt Reactor MOV Board 2C

1. Loads for Unit 2 that are comparable with those above.

Table IV (From Section 8 of the FSAR)
250-VOLT d-c CONNECTED LOADS

Unit Battery Board:

1. Reactor Motor-Operated Valve Boards (3 per unit, A, B, and C)
2. Turbine Building Distribution Board (1 per unit)
3. Circuit Breaker Board 9-9 (1 per unit)
4. 480-Volt Shutdown Board Controls
5. Emergency d-c Lighting
36| 6. Unit Preferred a-c Motor Generator
7. 4160-Volt Shutdown Board Controls (unit 3)
45| 8. Bus-tie Board
9. Cooling Tower Switchgear

Station Battery Board

1. Main Turbine Emergency Bearing Oil Pumps
2. Main Generator Emergency Seal Oil Pumps
3. Plant Preferred a-c Motor Generator
4. D-c Emergency Lighting at Diesel Generator Building
5. Distribution Board 9-24
6. Diesel Generator Air Compressors
45| 7. Cooling Tower Switchgear

Reactor MOV Board Loads

1. Autodepressurization Relief Valves (MOV Boards A, B and C)
2. Pressure Relief Valves (MOV Boards A, B and C)
3. Main Steam Isolation Valves Solenoids (MOV Boards A and B)
4. Recirculation M-G Set Emergency Oil Pumps (MOV Boards A and B)
5. Backup Scram Valves (MOV Boards A, B and C)
6. RHR Shutdown Isolation Valves (MOV Board B)
7. Division I Engineered Safeguards Logic Power Supply (MOV Board B)
8. Reactor Building Emergency d-c Lighting (MOV Board B)
9. HPCI Turbine Controls and Auxiliaries (MOV Board A)
10. HPCI Valves (MOV Board A)
11. RCIC Turbine Controls & Auxiliaries (MOV Board C)
12. RCIC Valves (MOV Board C)
13. Division II Engineered Safeguards Logic Power Supply (MOV Board A)

35|250-Volt d-c Control Power Supplies

45| 1. 4160-volt Shutdown Board Controls (units 1 & 2)

Auxiliary Power Supplies and Bus Transfer Schemes — Table V (From Section 8 of the FSAR)

Board and/or Main Bus	Power Sources Normal	Power Sources Alternate	Remarks
4160-V Start bd 2 — Start Bus 2D	COM SS TR B, secondary 2, fed from Athens or Trinity 161 kV lines	COM SS TR A, secondary 2, fed from Athens or Trinity 161 kV lines	
4160-V Bus Tie Board	Cooling Tower Transf TCT1	Cooling Tower Transf TCT2	Manual transfer from the normal power source to the alternate power source, or vice versa, is provided by operator; breakers [illegible] and [illegible] from the cooling tower control board or units 1 & 2 control room, control switches for these breakers are also provided on the 4160 v cooling tower switchgear A.

Board and/or Main Bus	Power Sources Normal	Power Sources Alternate 1	Power Sources Alternate 2	Power Sources Alternate 3	Power Sources Alternate 4	Remarks
Shutdown Bus 1 (4160-V)	4 kV unit bd 1A or 2D pre-selected on-line unit, fed from secondary 1 of unit SS TR	4 kV unit bd 2B or 1A (that source not pre-selected for "normal")	Same 4 kV unit bd of pre-selected unit, but fed from start bus 1 A or 1 B	Two diesel generators, if required for backfeeding a pre-selected 4 kV unit bd (1A, 2D) See also remarks for items 13, 14, 15, and 16	Bus tie board	The two independent shutdown buses normally supply 4160-V power to assigned 4160-V shutdown boards, with each bus serving as the normal source to two boards and as the alternate source to the two other boards. Of the two possible feeders to each shutdown bus from the two 4 kV unit boards, one feeder is pre-selected manually as the normal source to that bus. Automatic delayed transfer from the normal to an alternate 1 source is initiated by undervoltage on the normal source. Automatic instantaneous transfer from the normal to an alternate 1 source is initiated when the normal source 4 kV unit board normal source breaker trips. If an alternate 1 source is not available, the transfer is prevented, and the normal source becomes alternate 2 source. Automatic transfer is initiated after time delay or the presence of an accident signal. Alternate 3 and 4 sources may be selected manually only.
Shutdown Bus 2 (4160-V)	4 kV unit bd 1B or 2A, of pre-selected on-line unit, fed from secondary 1 of units SS TR	4 kV unit bd 2A or 1B (that source not pre-selected for "normal")	Same 4 kV unit bd of pre-selected unit, but fed from start bus 1 A or 1B	Two diesel generators, if required for backfeeding a pre-selected 4 kV unit bd (1B, 2A) See also remarks for items 13, 14, 15, and 16	Bus tie board	

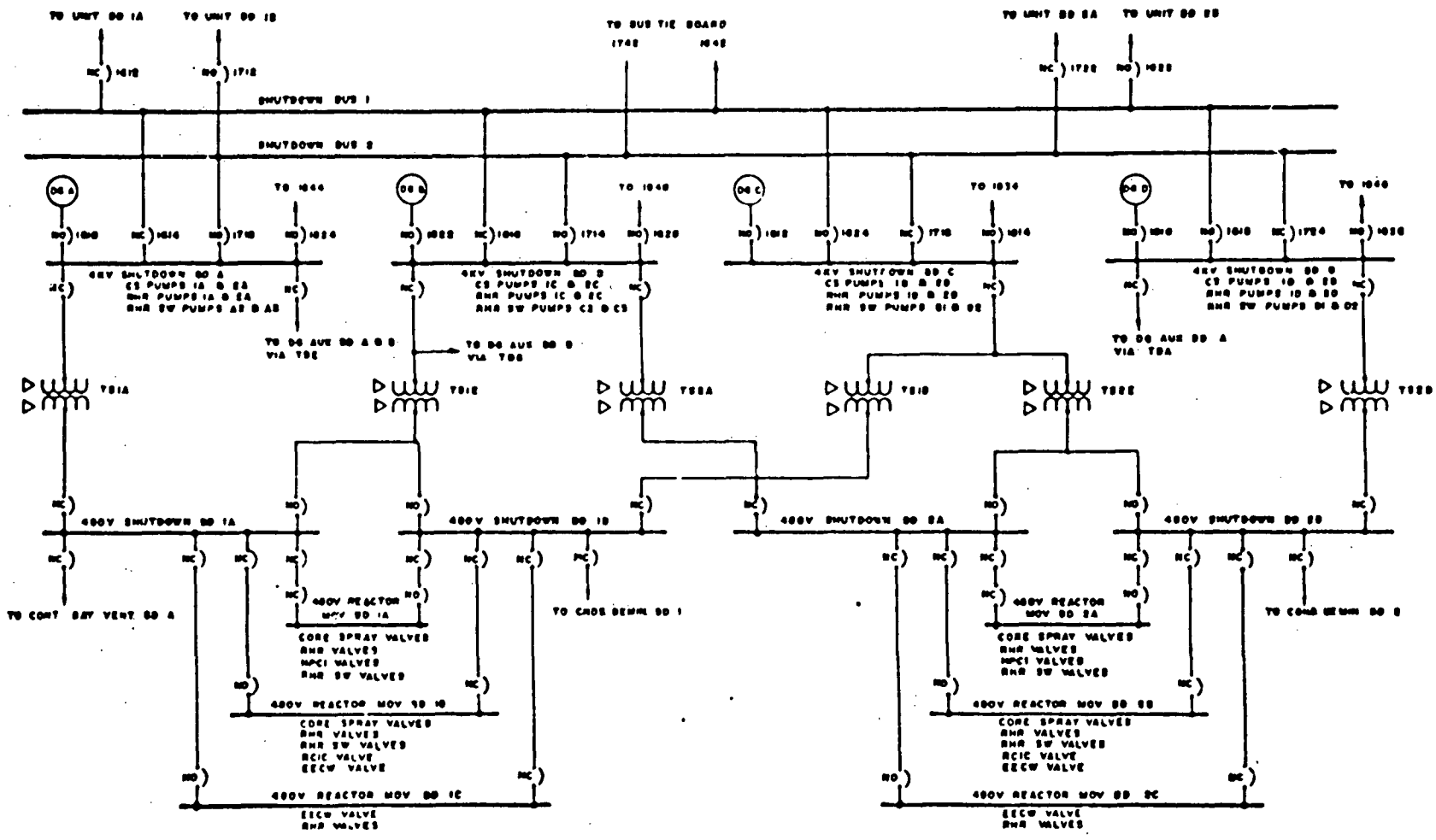
Table V (From Section 8 of the FSAR)

Auxiliary Power Supplies and Bus Transfer Schemes -

Board and/or Main Bus	Power Sources: Normal	Alternate 1	Alternate 2	Alternate 3	Remarks
4 kV Shutdown Board A	Shutdown Bus 1	Shutdown Bus 2	Diesel generator A	Manual, access connection to diesel generator 3A	(See also remarks for Items 5 and 6.)
4 kV Shutdown Board B	Shutdown Bus 1	Shutdown Bus 2	Diesel generator B	Manual, access connection to diesel generator 3B	Automatic delayed transfer from the normal to alternate 1 source is initiated by undervoltage on the normal source, and automatic return is initiated by normal voltage on normal source. These transfers are blocked after time delay in the presence of an accident signal.
4 kV Shutdown Board C	Shutdown Bus 2	Shutdown Bus 1	Diesel generator C	Manual, access connection to diesel generator 3C	All diesel generators are automatically started by an accident signal, loss of start bus voltage (see item 1.d), or by loss of voltage on its shutdown board for 1.2 seconds after [illegible] loads without voltage on the shutdown board, all its breakers and all its loads except 4160-480-V transformers are automatically tripped. Alternate 2 source is then automatically connected. Manual return to the normal auxiliary power system is permitted if normal auxiliary power system voltage returns and if a unit is not in early stage of accident.
4 kV Shutdown Board D	Shutdown Bus 2	Shutdown Bus 1	Diesel generator D	Manual, access connection to diesel generator 3D	

Auxiliary Power Supplies and Bus Transfer Schemes - Table V (From Section 8 of the FSAR)

Board and/or Main Bus	Power Sources Normal	Power Sources Alternate	Remarks
480-V Shutdown Boards			
Unit 1, 480-V Shutdown Bd 1A	4 kV shutdown bd A	4 kV shutdown bd B	Transfer from the normal to the alternate source is manual. Interlocking is provided to prevent manually transferring to a faulted board and to prevent paralleling two sources.
Unit 1, 480-V Shutdown Bd 1B	4 kV shutdown bd C	4 kV shutdown bd B	
Unit 2, 480-V Shutdown Bd 2A	4 kV shutdown bd B	4 kV shutdown bd C	
Unit 2, 480-V shutdown Bd 2B	4 kV shutdown bd D	4 kV shutdown bd C	
Unit 3, 480-V Shutdown Bd 3A	4 kV shutdown bd C	4 kV shutdown bd D	
Unit 3, 480-V Shutdown Bd 3B	4 kV shutdown bd A	4 kV shutdown bd D	
480-V Reactor MOV Boards			
Unit 1, 480-V Reac MOV Bd 1A	480-V Shutdown Bd 1A	480-V Shutdown Bd 1B	
Unit 1, 480-V Reac MOV Bd 1B	480-V Shutdown Bd 1B	480-V Shutdown Bd 1A	
Unit 1, 480-V Reac MOV Bd 1C	480-V Shutdown Bd 1B	480-V Shutdown Bd 1A	Automatic transfer from the normal to the alternate source is initiated by time-undervoltage on the normal source. Return to the normal source is automatic upon return of voltage to the normal source.
Unit 2, 480-V Reac MOV Bd 2A	480-V Shutdown Bd 2A	480-V Shutdown Bd 2B	Transfer from the normal to the alternate source is manual. Interlocks prevent transferring a fault from one source to another and paralleling sources.
Unit 2, 480-V Reac MOV Bd 2B	480-V Shutdown Bd 2B	480-V Shutdown Bd 2A	
Unit 2, 480-V Reac MOV Bd 2C	480-V Shutdown Bd 2B	480-V Shutdown Bd 2A	Automatic transfer from the normal to the alternate source is initiated by time-undervoltage on the normal source. Return to the normal source is automatic upon [illegible]



SHUTDOWN AUXILIARY POWER

EXHIBIT C-2 FIGURE 1

Unit 1 Systems Lost During Fire

AUTOMATIC DEPRESSURIZATION SYSTEMS	PROBABLE CAUSE OF LOSS
7 of 11 Relief Valves Inoperable From Control Room	Loss of 250 VDC Boards 1A and 1B
Manual Control of 4 Remaining Valves	Lost in Supply to the Valves

Therefore, manual operation of relief valves lost on Unit 1. Automatic pressure relief still available. Air stored in the valve accumulators provide a discrete number of operations after the loss of the normal air supply.

HIGH PRESSURE COOLANT INJECTION SYSTEM	PROBABLE CAUSE OF LOSS
HPCI	Lost because power to valves lost on 250VDC MOV Board 1A and Power to steam isolation valve lost on 480V MOV Board 1A.

RESIDUAL HEAT REMOVAL SYSTEMS	PROBABLE CAUSE OF LOSS
RHR System 1A	Lost because power to valves lost on 480V Board 1A.
RHR System 1B	Lost because power to valves lost on 480V MOV Board 1B and loss of power to pump motors on 4KV Shutdown Board C.
RHR System 1C	Lost because power to valves lost on 480V MOV Board 1A.
RHR System 1D	Lost because power to valves lost on 480V MOV Board 1B.

Therefore, all RHR systems lost on Unit No. 1.

CORE SPRAY SYSTEMS	PROBABLE CAUSE OF LOSS
Core Spray System 1A	Lost because power to valves lost on 480V MOV Board 1A.
Core Spray System 1B	Lost because power to valves lost on 480V MOV Board 1B and power to pump motors lost on 4KV Shutdown Board C.
Core Spray System 1C	Lost because power to valves lost on 480V MOV Board 1A.
Core Spray System 1D	Lost because power to valves lost on 480V MOV Board 1B.

Therefore, all core spray systems lost on Unit 1.

STANDBY LIQUID CONTROL SYSTEM	PROBABLE CAUSE OF LOSS
SLC Pump 1A plus squibb valves.	Lost because power to pump motor and valves lost on 480 V Shutdown 1A.
SLC Pump 1B plus squibb valves.	Lost because power to pump motor and valves lost on 480V Shutdown Board 1B.

Therefore, SLC lost on Unit 1.

REACTOR CORE ISOLATION COOLING SYSTEM	PROBABLE CAUSE OF LOSS
RCIC	Power to the steam isolation valve lost on 480V MOV Board 1B.

REACTOR PROTECTION SYSTEM	PROBABLE CAUSE OF LOSS
RPS MG Set 1A	Lost because power to motor lost on 480V MOV Board 1A.
RPS MG Set 1B	Lost because power to motor lost on 480V MOV Board 1B.
Units 1 and 2 Standby Coolant Valve	Lost because power test valve lost on 480V MOV Board 1B.

TEMPORARY CABLE LOG

Cable/Wire Number	From	To	Function
1 PL992	480V R. Vent Board 1A	480V T. Vent Board 1A	TB Exhaust Fan 1A Control
1 PL1002	480V R. Vent Board 1A	480V T. Vent Board 1A	TB Exhaust Fan 1B Control
1 PL1187	480V R. Vent	480V T. Vent	TB Exhaust Fan 1B Control
ES 1607-IA	Pnl 25-45A	Pnl 25-45C	Common accident signal to diesel generator
ES 4107-IIC	Pnl 25-45A	Pnl 25-45C	Common accident signal to diesel generator
ES 4393-IIC	Pnl 25-47C	4KV SD Bd C	Diesel Generator C Differential Circuit
ES 4457-IIC	Pnl 25-47C	4KV SD Bd C	4KV Breaker 1812 Close Circuit
ES 4285-IIC	Pnl 25-47C	4KV SD Bd C	Diesel Generator C Protective relay circuit
1ES 4382-II	Pnl 9-36 Unit 1	Pnl 25-219 Unit 1	Plant refuel zone isolation signal
1ES 4385-II	Pnl 9-36 Unit 1	Pnl 25-221 Unit 1	Reactor zone isolation initiation
1ES 4375-II	Pnl 9-36 Unit 1	JB2801	FSV-64-40 Control Reactor Zone exhaust
1ES 4330-II	Pnl 9-36 Unit 1	480V Dsl. Aux. Board B	Relay LRB-6 Control
1ES 4368-II	Pnl 9-36 Unit 1	JB2788	FSV 64-44 Control refuel zone exhaust
1ES 4353-II	Pnl 9-36	JB2801	FSV 64-43 Control Reactor zone exhaust
1B76	250V R. MOV Board 1A	Battery Bd. 1 Pnl 2	250 VDC Reactor MOV. Board 1A Normal Feed

EXHIBIT C4
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Cable/Wire Number	From	To	Function
1B77	250V R. MOV Board 1A	Battery Board 2	250 VDC Reactor MOV Board 1A Emergency Feed
1A2318	Pnl 9-4	JB 3404	Seismic Alarm
1A2319	Pnl 9-4	JB 3404	Seismic Alarm
1A2320	Pnl 9-4	JB 3404	Seismic Alarm
1ES4331-2	Pnl 9-36	JB 2567	Reactor building vent
1R1671	Pnl 9-18	Pnl 25-59	RHR Sys. I Discharge Press Inst. loop 74-51
1R1691	Pnl 9-19	Pnl 25-62	RHR Sys. II Discharge Press Inst. loop 74-65
1R1578	Pnl 9-18	Pnl 25-1	CS Sys. I Press Inst. loop 75-20
1R1588	Pnl 9-19	Pnl 25-60	CS Sys. II Press Inst. loop 75-48
1R1000	Pnl 9-5	Pnl 25-18	CRD Cooling Water 4P Inst. loop 85-18
1R1031	Pnl 9-19	Pnl 25-18	CRD Cooling Water Hdr. Flow Inst. loop 85-25
1R1032	Pnl 9-19	Pnl 25-18	CRD Cooling Water Hdr. Flow Inst. loop 85-25
1CR950	JB 1043	Computer	Vessel Temperature TE 56-1
1CR955	JB 1043	Computer	Vessel Temperature TE 56-3
1CR960	JB 1043	Computer	Vessel Temperature TE 56-17
1CR965	JB 1043	Computer	Vessel Temperature TE 56-30
1R1681	Pnl 25-62 Unit 1	Pnl 9-19 Unit 1	RHR Instrumentation
1R1686	Pnl 2562 Unit 1	Pnl 9-19 Unit 1	RHR Instrumentation
1R1656	Pnl 25-59 Unit 1	Pnl 9-18 Unit 1	RHR Instrumentation

Cable/Wire Number	From	To	Function
1R1666	Pnl 25-59 Unit 1	Pnl 9-18 Unit 1	RHR Instrumentation
1R2740	Pnl 9-3 Unit 1	Penet. CB RS E1 565	Drywell Temp. Recorder
1NM2-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A Signal
1NM3-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A HV
1NM4-IA	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM A -15 VDC
1NM18-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B Signal
1NM19-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B HV
1NM20-IB	Pnl 25-27 Unit 1	Pnl 9-12 Unit 1	SRM B -15 VDC
1NM10-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C Signal
1NM11-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C HV
1NM12-IIA	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM C -15 VDC
1NM27-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D Signal
1NM28-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D HV
1NM29-IIB	Pnl 25-61 Unit 1	Pnl 9-12 Unit 1	SRM D -15 VDC
1PC518	250V R. MOV Board 1B	Pnl 9-43 Unit 1	FCV 69-2 Control
1PC517	250V R. MOV Board 1B	JB 1088 SR4 B1 593	FCV 69-2 Control
1PC516	250V R. MOV Board 1B	FCV 69-2 Motor	FCV 69-2 Power

Cable/Wire Number	From	To	Function
1PC506	480V R. MOV Board 1B	Pnl 9-43 Unit 1	FCV 69-12 Control
1PC505	480V R. MOV Board 1B	JB 977	FCV 69-12 Control
1PC504	480V R. MOV Board 1B	FCV 69-12 Motor	FCV 69-12 Power
1V2173	480V R. MOV Board 1C	Pnl 9-4 Unit 1	FCV 69-16 Control
1V2171	480V R. MOV Board 1C	JB 1031	FCV 69-16 Control
1V2170	480V R. MOV Board 1C	FCV 69-16 Motor	FCV 69-16 Power

Calhoun calling Green. Fred Chandler called and said that somebody from BFNP had asked him to get some people out to do a little checking on what cables go in certain cable trays and all.

What I have going is a kind of a task group of Construction and DPP (mostly Construction) looking toward the repair of this and my immediate representative is Jerrold Dewease working with Ben Gant and they have lined out a plan of attack toward repairing this and Jerry just went over with me the plan and the design part was to get Design to identify everything in these trays, so they can have this information ready when we went to taking out circuits to chop them out to place in new circuits. So I think what we are talking about is not just today's problem but toward getting these units back.

Okay. Fred's concern was that they have not gone through you.

But they have gone through me and I've got a representative on this task force just assuming we are going to call on Construction to help to get the unit back.

Okay.

Here's Jerry's notes. It starts out, list all trays that show damage. DEC will complete a list of cables on each tray. DEC will identify where they go and this sort of thing and then the power and control cables can be spliced, the thermocouples coax will be replaced. DED should be notified and I understand they have been. (JRC says yes.) That's what it is--just trying to get together with Construction and looking a little bit ahead.

Okay, Fred wanted to relay this back through here so we wouldn't get fouled up, but I think I can tell him on this particular problem to work directly with that task force. Right?

Yes, work with the task force and I believe Ben Gant is the construction man that is taking the lead. But, of course, Ben might have some other engineers involved, too.

Yes. Now hold on, I've got some other messages for you too.

Recford Arnold and Fred Von Hollen are coming down there to help you.

Okay.

Mr. Thomas has something here. (EFT speaks in background--JRC repeats EFT's question) Is # 1 turbine safely on turning gear?

Yes, they tell me it sounds good.

No troubles with bearings as far as they can tell?

No.

Okay, now Frank Long, Atlanta, called and says he is sending Bob Sullivan, Bill Little, and Bower (electrical man). So they left maybe thirty minutes ago and they will be down there sometime tonight.

Okay.

He asked me did you know whether your control rods were in on unit 1 and I told him that I didn't know.

I've got indication that they are all in.

Okay, good. Now what's the situation on the fire?

Jack, we're fighting it with water now and our men where breathing air is a big thing, but they have gone in twice and had it knocked down with water and had to get out and each time they go back in_____and they are in there for another entry right now and I've got hopes that the fire is about under control.

So, they are not using too much water apparently.

No, they are not trying to use too much water.

Do you have any indication of how much of that cable tray outside is involved?

Yes, there is a vertical that runs up and down the wall and two places on horizontal and I din't know just how much, but I got the thing in my mind probably about 15 feet horizontal.

Okay, that's the sort of information we wanted. Now, has there been any change in the torus temperature on unit 1?

No, right now unit 1 has _____. I've lost my unit valves on unit 1, and I know what it is and I know what to do about it. They have burned through an air line, so killed the air and we killed control air while relief valves went shut and I'm trying to establish vacuum and going to use the steamline drain stub to help pull steam off the turbine and I don't know what the absolute torus temperature is. I've been down watching the firefighting efforts, Jack. Just came up.

But, you've got a way that you'll be able to quit but you're not dumping anything in the torus anyway, are you?

Not right now, don't have relief valves. I'll get another rundown here shortly and get back at you.

Yeh, let us know as soon as possible.

Okay.

(Later) Okay, I am reporting that the fire appears to be out. Right now, unit 2 is still down at about 100 pounds and makeup water is the hotwell pump. Okay, unit 1 I've got in a condition where I'm leading steam to the condenser through the drain lines on the steam lines and hold the vacuum with the hoggers and making up water with the control rod drives and I seem to be holding pretty steady at about 500 pounds. The question you asked me: unit 2 torus temperature is 106° and unit 1 torus temperature is unknown.

Okay, you don't have any way to get down to the torus, do you?

I haven't attempted it yet, I have too many other problems.

Okay. Alright. Jim, I guess I didn't have anything to talk to you about. Oh, we've got 35 more breathing air bottles on the way down to you.

Okay, I don't know which came in where, but we got some air in here from some place that we are using to recharge cylinders and we are also using Chemox and that's working pretty well. I got a call that Sullivan, Little, and some other NRC inspector are traveling tonight and will get here some time tonight so all our problems will be over. (JRC laughs).

They will square you away, I am sure.

We probably have a violation. We've kept very poor logs. (JRC laughs)

No doubt.

I wouldn't say we are out of the woods, but I feel better now that the fire is out and we should be able to get in the reactor building in a little while without all this breathing air and mask, you know.

Can you expand a little bit on that air line that burned in two?

Just a minute. Jack, I can't give you the definitive. I'm talking to Joe Mantooth, who was the operator in there, and he says that in that area there is a 6" riser going up. He doesn't think it was the main riser itself, but probably a leg off of it, but that there was an air blow in there blowing into the fire. Keep in mind that it was very poor visibility. With flashlights just a matter of several feet so he's not definitive of it all.

Okay, thank you.

Okay.

Jim has reported that he has the fire out.

Okay.

What was taking him so long, he finally resorted to water, but he wanted to do it with the spray and just controlled amounts of it and what he would do was to put a little bit on it and wait a while and it would flare up and he would put some more on it. Now apparently this is in a space of about 15 feet in this particular cable tray. So he has the fire out. Unit 2 is holding at 100# pressure. And he is feeding water to unit 2 with the hotwell pumps. No problem on unit 2. Now unit 1 since I called you last ran into another problem. They had an air line that was in the vicinity of this fire and they lost air pressure to the relief valves so his relief valves went out on him and now they have the heatsink as the main condenser on unit 1 and taking steam through the main steam drain lines to the main condenser on unit 1. This seems to be stablizing the pressure on unit 1 to 500 psig. Unit 2 torus temperature is 106° F. Unit 1 torus temperature is unknown.

What happened there?

Well, he lost his unit 1.

Measuring that directly before estimating?

He indicated to me at one time that he was getting some indication in the auxiliary shutdown room or on the emergency control room that he got a reading there and that was the one I gave you at what 126°.

On what #1?

Yeh, 126° and I'm pretty sure he told me he was getting that from the emergency control room but now he says he doesn't know what the temperature is on the torus. So, I'm a little bit unclear as to where he got that first reading.

Okay, that just could have been some good educated guess.

I'm pretty sure that he told me he got it from the control room. But I'm not absolutely sure.

Okay, what's the status of main control room? Is that all filled with smoke still?

Oh, no. It never had much smoke in it. The main control room is perfectly okay. They had some smoke coming through a hole close to the control deck in the center of the room. They stuffed some rags into it and got it stopped. But he thinks he will have the smoke out of the reactor building pretty soon. We got a call through some of our people that Mr. Ed Giller was calling from Washington. Do you know Ed Giller? I probably figured that he was their duty man.

Yeh, he's probably their duty officer. Several of them were in the offices up there. Just to see what was going on.

Now, we ended up by asking him to call you. Is that okay?

Yeh. We have some people up there in our headquarters offices. The doggone public news media types will probably drive you out of your mind. Okay, your people did put out a news press release?

Yeh, we put one out about 4:30. Somewhere close to 4:30.

What kind of fire fighting--Did you have any problems with your station fire system?

As far as I know, it worked fine. The CO_2 system in that particular room works okay. But it ends up with a cable fire like this the CO_2 won't put it completely out. Sort of like a mattress smoldering.

Yeh, you can't really get to cool it.

Right? No cooling. So it prevents a conflagration but it won't just finally put it out. Now the Athens fire department got there quickly and I guess I told you we locked them right in to the reactor building with fire trucks. So they were there and acted as advisors and did real well.

Oh, you didn't have to use their equipment?

Yeh, we used some of their equipment too. Principally, this particular (now I'm guessing here now) but Jim was going to use his spray nozzle for this water and I assume that he did use the Athens fire department spray nozzles. So that part of the plan really worked out well. He had the fire department there before I ever got here in the emergency control center here in Chattanooga.

Is that part of your regular fire plan?

That's part of the regular fire plan. See, Jim takes the Athens fire department into the plant and trains them in radiation and also takes them through the plant so they are familiar with the inside of the plant, too. This is all a part of the plan.

Tell me what you know about the cooling thing for unit 1. Give me that again _ _ How you are keeping that temperature.

We are taking steam through the drain lines. These are fairly small drain lines that go to the main condenser. And we're passing steam through those drain lines to the main condenser. And maintaining a vacuum on the main condenser with the hogging pumps.

Do these drain lines have that capacity to take care of that for you?

A 2" drain line off each main steam line. Yeh. So it's holding and it will get better as time goes on. As the decay heat drops.

Does each of your units have two condenser circulating cooling pumps?

There's 3 main circulating condenser pumps. Yeh. Each unit.

Condensate pumps, you've got two or three?

Three, six I guess condensate booster pumps. One for each condenser--Three condensate boosters.

On each unit?

Yeh.

That explains. I was trying to figure whether there was one or two on a plant. You're relying on them. That handled it pretty well, didn't it?

Oh, yeh, we had plenty of cooling water.

It looks like everything is under control.

Things are looking up.

That fire wasn't as extensive as it appeared to be? Pretty localized by 15-foot area length?

The 15-foot area I am talking about is outside the spreading room. The fire went through a penetration the way it looks so I don't know what to say about inside the spreading room.

That was probably pretty extensive in there.

Well, we just don't know. It must have been to affect two units. But we have a feeling now it is nothing like completely wiping out the spreading room. We can say maybe three or four cable trays in there. Just some guess work.

Really don't know?

Really don't know.

Only thing we can say right now is that it could have been a hell of a lot worse.

Oh, yeh.

You know, when you talk about a fire in the spreading room, you've really got problems.

It would affect just about everything.

Yeh, you know everything for those two units come through that one room. It's common to both units, just like the control room is common to both units.

That sorta shoots your redundancy.

We're got a lot of questions to ask about this.

Well, I hope it turns out that it wasn't really as extensive as it seems to have been.

Smoke really fools one and he just doesn't know. Maybe before the night is over, we will have a pretty good feel for the extent and all because he is getting the smoke cleared out now.

Okay. I'll just sit back and wait. I sure do appreciate your calling me, Jack.

Thank you, Frank. Bye.

Fire is essentially out. On the spreading room end, every now and then you see a glow and we got a man standing by like continuously. Now we have set up routine patrol in the drywell and the area where we have fire, you know. Now, #1 reactor building is cleared by the health physicist, but I haven't got it cleared by the industrial hygienist. They are concerned about the CO. So I am still in a mask. Unit 2 is in the best shape and right now we are attempting to get it on shutdown cooling. And it looks like we've lost the (I guess it would be the A & B shutdown) boards because we have lost the A and C pumps, this type of thing you know. And if you'll think that one through, it means that what we have really lost are the controls because where the fire was there were no power cables.

Yeh, okay.

So unit 2 has two RHR pumps, two core spray pumps, what have you. Unit 1 we just got a relief valve open. I think the last time I told you we couldn't get air on a relief valve and we have just got in a relief valve so the pressure is coming down.

Okay.

At the present time I don't have any way to get one in unit 1 except the control rod drives for when I get down to 150 # I can get it in through the condensate system.

Yeh, I understand.

Unit 1: One thing I am going to do is to put in the spool pieces on the RCIC. I think it would be inoperable except the steam isolation valves inside the containment is shut and if I get spool pieces in, it will give me another way to put water using auxiliary steam.

From heating boiler?

Yeh. When I get operators down to it, I can open the valves manually.

Yeh.

And we are starting from nothing to make a procedure to flush the shutdown cooling _____ and put in service on unit 1. There will be a manual alignment, you see. On unit 1 I have most of the CAM's back in service and I have Construction here, and what we plan to do was to take out a component at a time starting with the _____ well, I'm trying to find a way to get core spray and RHR and this thing back in service, you know. And starting with the _____ I'm going to start like a component at a time and say well I need this valve. Ben Gant's people will work with us and we're going to disconnect the control cables from that thing and at the breakers I will put in a local pushbutton that I can operate that valve with and starting a scheme of this nature to try to wind up to get some of this safeguards equipment operable for unit 1 for local control. Now we have established that from here on we are not going to do any of these bastard things without a work plan.

That's good.

We've done some things out of the extreme. We are trying to reestablish control. And Studdard's people are working with Ben Gant's people to identify each of the components that we need and I think we have and could operate this core spray and they are making the proposal what they think they can do to get it in service since the boards hot and apparently all we have lost are controls for it and then we will plan it through. And it looks like most of the night is going to be spent in making procedures to get this thing on shutdown cooling and starting on reviewing procedures and workplans toward getting some of the safeguards equipment back in service.

Okay, Jim, do you know the temperature in the drywell or the torus temperature?

Unit 1 the drywell temperature? I don't know the pressure, I don't know the torus temperature, I don't know the torus level. And we are giving a worklist of priorities to the instrument people and one of the (understand several jobs can run simultaneously) but one of the top priority jobs is that we are going into the drywell and set up a local neutron monitoring station (not in the drywell) in the reactor building _____ set a local neutron monitoring station and we've established that when we set that up we will have to keep an operator in to collect data all the time and then about the No. 2 job will be torus level. This is of particular concern because we don't have any idea where the level is. The side torus temperature is a low priority getting the permanent instrumentation in that will _____ the thermometer to the side of the torus.

Sure.

Somewhere down the list appears then drywell temperature and drywell pressure. We have the small vent open on the drywell that is, the 2" vent; so we know that we are probably not in any overpressure condition or anything like that. And that does vent out through standby gas, you understand that.

Yeh.

Drywell pressure, I feel sure, is probably nothing. But I don't know temperature.

Okay.

_____. A truck with air just came through the gate.

Okay, good, that should get us out of a bind some on air. I just sent Athens fire department home. And I don't how the hell we ever pay them or anything like that.

Well, they did a good job.

They did us a good job. And they left some of their equipment here that they couldn't find and what have you and I said we'll find it and get it to you or we'll pay you for it.

Okay, sure.

And the only other thing I've been kind of treating them by buying meals. I've bought meals for Athens firemen and everything else.
You're in trouble.
Probably am. That's the biggest _____ decision, you know. And blindly ordered 50 breakfasts.

Jim, unit 1 reactor water level, what's the water level?

I don't know, but there is no problem with it.

Okay. Atlanta asks that ever once in a while.

Well, we haven't had a level problem. Unit 1 we didn't have any way to get steam out so you didn't have to put water in.

Right.

And it's not that _____ humorous. That situation, unit 1 right now is still not good. The only way we are putting water in is control rod drive. The bright spot _____ we did get a relief valve open and we still do have a condensate system to put water in when we get pressure down.

So the biggest problem is getting cooling to the torus. Isn't it?

Right.

The crosstie doesn't help you any in this case, does it?

No, crosstie don't help any in this case.

Okay, Jim, what are you going to do for yourself?

Well, I'm going to stay here tonight. I'm going to sleep here.

Alright.

Either Studdard or Hunkapillar are going to be here all night.

Allan Qualls is down there some place if he can help you any. Probably in Athens at home.

Oh, hell, Allan Qualls is probably down here visiting his daughter.

Yeh.

It's good to know. If we need him, we will call him. Don't think we are in too bad a shape at that level in that we got some pretty competent shift engineers, you know. If we run into any problem, since we had so many out here today, tat it is going to be a damn long night for them. Mainly, we are going to keep maintenance crew and all crafts here all night long. And Construction has a good many engineers and electricians and craftsmen out tonight just standing by to help us. They have cancelled all construction work, but they did bring in a crew to stand by for any support we need.

Is there anything that we need to do for you in the morning that we could get started tonight?

Jack, the main thing that came to mind that you can do for me and I'm thinking _____ by April 1 I'm supposed to have this fellow from ABC to take a film report of the plant, you know, some of these tour guide groups, if we can get some of those cancelled out.

That was April 1?

Right.

Well, okay.

That's a little facetious at this time. We can worry about that Monday. But right now I don't know of the support I need from you.

Okay.

You can cancel the relief valve test on unit 2 for awhile.

Yeh, we'll do that.

Jim, Fred Von Hollen and Reeford Arnold, I don't guess they have arrived yet, but they are on their way and if you need any more people out of us in the morning let us know.

Fred Von Hollen and Reeford Arnold are on the way down for any help they can be.

We will keep someone here in the office all night.

Somebody is going to be at this number all night.

Right.

I'll call you if I need you, but I don't know of a specific that you can do for us right now.

Well, I would like to know when you get the RHR restored. If it is sometime during the night, let them know here in the office

On unit 1?

Yeh.

If we get an RHR, I'll let you know.

When you get cooling to the torus. Yeh.

Okay.

Okay.

Log of J. K. Calhoun, Chief, Nuclear Generation Branch, TVA, for March 22, 1975.

J. K. CALHOUN LOG

MARCH 22, 1975

2:45 p.m. CDT - JAC called JRC

2:40 - JRC talked to HJG

2:41 - Started calling CEOC (Called JCHolmes at 2:45 - not home; called JAC at 2:45; and called EFT at 2:47)

2:50 - Left for CEOC office

3:10 - Arrived in CEOC office

3:20 - Called CJHodges

3:15 - RCPrice arrived

3:23 - RCPrice called TVA Information Office

3:23 - Called GPStone - no answer

3:25 - Chris Eckl going to Information Office in Knoxville

3:26 - CCFussell says he is on way to CEOC office

3:27 - CAMcLaughlin and JAC reported to Division Emergency Center

3:35 - JAOppold reported to take charge of CEOC

3:40 - JEWatson notified.

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3:40 - HJGreen reported following information:

Unit 1 - 230 psig + 55 waterlevel in reactor; condensate booster pump in use supplying water to reactor

Unit 2 - + 35" waterlevel; 150 psig boiler pressure; RCIC being used; backup feedwater system is the hotwell pump

Unit 1 has no ECCS valve control

Reactor building has no area radiation monitoring

Apparently fire is in cable tray (TE) in middle of spreading room wall between reactor building and spreader room

Smoke so heavy can't see

Unit 1 torus water up to 126° F.

No torus instrumentation

100 meals ordered for 6 p.m.

Breathing air may be needed

Locking fire truck into building

Unit 2 is on turning gear

Unit 1 sitting - No lift pumps and not on turning gear

News release made to Chris Eckl

4:04 - HJG

Unit 1 is in worst shape - Has a reactor water level of + 55 and 200 psig pressure

Condensate booster pump in service

Unit 1 does not have torus indication

Torus temperature is 126° F.

Unit 1 does not have nuclear instrumentation

Unit 1 does not have core spray, RHR, HPCI, RCIC, etc., probably because MOV boards are dead

Unit 1 has no turning gear or lift pumps

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Unit 1 drywell temperature 330° F. or 250° F. - Don't know which because indication is erratic.

Pressure in drywell is 2.2 psig

Only have two drywell cooling fans out of ten

Unit 2 has some core spray, some RHR, HPCI, RCIC, and torus cooling

Unit 2 level in torus is 4"

Unit 2 torus temperature is 130° F. and dropping

Unit 2 drywell temperature is 195° F.

Unit 2 turbine is on turning gear

Unit 2 has neutron monitoring

The nursing station is manned - Nobody has been hurt.

4:23 - Tom Fullner, BFNP storeroom, called

Can't get Selox Company in Chattanooga - BFNP needs breathing air

We will get in touch with Selox and ask them to send some breathing air to BFNP

50 cylinders breathing air on TVA Contract No. 70-X53-BPA 58

4:33 - HJGreen

Service Shops leaving with fire extinguishers and an air compressor

Fire is out on spreading room side

Foam was used in pulling cables - Wasn't fireproof and acting as blow torch

Athens fire department can charge our air bottles at BFNP

At 4:30 both turbines on turning gear

Liquid CO_2 truck being ordered because Cardox system is getting low

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4:55 - Marvin Mathis on way from Chattanooga to help inspect turbine for possible bearing damage

4:58 - Selox Company, Chattanooga, is sending 15 bottles breathing air to BFNP

4:59 - Operator says to call Mr. Frank Long in Atlanta (404) 633-2426

5:10 - HJGreen wants to decide what to use on fire

Determined to be "A"-type fire - He is being advised to use water on fire

Cable trays MCES2, MD, TE, and VE were involved in fire

Two more immediately near these were also involved

5:20 - Told HJGreen okay to use water on fire but use water under controlled conditions

Norman Moseley in Atlanta (404) 633-4012

5:40 - Fred Von Hollen and Reeford Arnold going to BFNP - Told to do so by RCPrice

6:06 - JDGlover

Enough heat has been generated in cable trays to cause flare up of fire after being put out

Having trouble with reactor building exhaust fans on unit 1

6:09 - HJGreen says hasn't used water yet. Fire keeps coming back from heat but he wants to hold out for another 30 minutes before using water.

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Cable trays extend into reactor building in a vertical and a horizontal direction

Jim says construction "candled penetration and it flashed" while sealing a penetration. Price has written statement from the men.

6:15 - Fifteen breathing air tanks on way from Chattanooga

6:22 - Fred Chandler's people say they are getting people in to check cable runs

6:29 - Frank Long, Atlanta sending Bob Sullivan and Bill Little and Bower to the plant

6:50 - HJGreen - Has created a task force to determine the cables involved in the fire and a course of action

All control rods on both reactors

Now fighting fire with water

Lost relief valves on unit 1 because of control air line failure in vicinity of fire

7:00 - Thirty-five more breathing air bottles on way to BFNP

7:15 - JEWatson called to get more information on problem

7:25 - Godwin Williams says Washington calling him

Ray Sullivan, ERDA, Oak Ridge says Ed Giller in Washington wants to know what happened. Told him to tell Giller to call Frank Long, Atlanta. We are keeping Mr. Long informed

7:47 - HJGreen - Fire appears to be out. Unit 2 reactor at 100 psig. Feedwater is from hotwell pump.

Unit 1 bleeding steam to condenser thru main steam drain lines. Holding at 500 psig.

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Unit 1 using control rod drive pumps for feedwater

Unit 2 torus 106° F.

Unit 1 torus temperature is unknown.

- Called Frank Long, Atlanta, and gave him a status report

- JACoffey called Fred Chandler to tell him that DED should talk directly to the cable task force on information regarding cable trays

- Charles McBride says Bob Dunnivan with the Huntsville Times heard that a workman was testing a penetration for air leakage with a candle.

- HJGreen says Milton Price is pretty sure that candle started fire. Planning on PORC meeting to decide work plan for tonight.

Green says cable trays outside spreading room burned for about 25 feet expanse maybe in 10 trays. Spreading room looks okay. Apparently, all fire outside spreading room. Signs of heat on floor in control room.

Pressure on unit 1 has started to climb

9:00 - Called Frank Long and told him about the candle possibly starting the fire

9:20 - Received following information from Henry Russell and Fred Chandler: Polyurethene or styrofoam sealant used to seal penetration after the polyurethene set ups, the workman checks with candle for leaks and then sprays with a fire proofing material. DED says workmen were pulling two new cables thru a penetration when the fire occurred

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9:30 - Henry Russell says the following cable trays were involved:

FM, MX, MD, VE, TE, FL

Only tray with engineered safeguards was MX.

Pulling cable 3E343302 from panel 9-36A, unit 3 auxiliary instrument room to 480-volt diesel auxiliary boiler B, panel 11A, elevation 583, diesel generator building

9:42 - Operator on unit 2 says relief valves are now open on unit 1 and the reactor pressure is dropping.

9:45 - EFThomas going home

9:46 - HJGreen says routine patrol established in area where fire occurred

Reactor building atmosphere pronounced okay by Health Physicist

on unit 1, A&B C&D shutdown boards apparently inoperative - controls probably are inoperative

Unit 2 has RHR's and other systems in service

Unit 2 has relief valves opened. Control rod drives are supplying water to unit 1 - 500 psig reactor pressure

Plan to put in spool pieces to connect the RCIC to use heating boiler steam

Unit 1 constant air monitors in service

Unit 1 water level okay

Unit 1 drywell temperature unknown; torus level unknown; torus temperature unknown

Small vent open on drywell containment/venting thru standby gas treatment

Breathing air arrived from Chattanooga

Athens fire department left plant

4/28/75

EXHIBIT D3

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10:15 - Gave status to Frank Long, NRC, Atlanta.

Frank says Washington wants following information:

Torus temperature and drywell temperature and drywell temperature time sequence when information is available

10:45 - Gave EFThomas latest status

11:00 - JRC left control cente. Willie Webb and George Messer will stay until 8 a.m. CECC members and others left about 10:15 p.m. Key members will congregate about 7 p.m. 3/23/75

4/22/75

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Summary of CECC Activities March 22-23, 1975

Larry S. Fox, Assistant to the Director of Power Production, 716 EB, Chattanooga
James A. Oppold, Assistant to the Director of Environmental Planning, 201 401B, Chattanooga
April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

In response to your request of April 26, I will provide your investigating committee with a few brief comments about the activities of the Central Emergency Control Center (CECC) on March 22-23. As you know, the CECC Director has the responsibility for evaluating, coordinating, and directing TVA's overall activities involved in coping with a radiological emergency. These responsibilities are set forth in the Radiological Emergency Plan (REP) which describes TVA policies, purposes, delegations, standards, and guidelines in order to protect TVA personnel, plants, and properties, as well as the health and welfare of the general public during a radiological emergency.

J. A. Coffey, Division of Power Production, notified me at 1605 on March 22, 1975, of the fire at Browns Ferry Nuclear Plant and informed me that the REP should be activated and the CECC staffed. I immediately informed E. A. Belvin, Environs Emergency Staff Director, and requested him to alert the State of Alabama. I arrived at the CECC in the Edney Building at about 1625 and learned that, as first alternate, I was to direct the CECC. The following division representatives were notified and asked to stand by on alert: H. G. Moore, Division of Environmental Planning; D. G. Powell, Division of Law; and W. B. Brown, Water Control Planning. Active participants were Dr. R. L. Craig, Division of Medical Services; C. E. Eckl, Information Office; C. F. McBride, Power Information Advisor, Office of Power; and Communication Coordinator, C. J. Hodges. I have drafts of a chronological log by J. A. Oppold, Dr. R. L. Craig, and C. J. Hodges.

The CECC notified the Nuclear Regulatory Commission in Atlanta of the Browns Ferry fire at 1703, and at 1712 a followup contact was made with the Alabama State Department of Public Health. Several telephone conversations were held with each agency throughout the night in an attempt to keep each apprised of the situation at Browns Ferry. This was possible since oral communications and frequent briefings were held with the Division of Power Production Emergency Center. At 2105 the Tennessee State Department of Public Health was notified as a public relations move rather than fulfilling a requirement of the REP.

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PAGE 1 of 3

BEST COPY AVAILABLE

Harry S. Fox
April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

Five areas of concern prevailed in the CECC on March 22:

1. Getting the fire out;

2. Maintaining control of the reactors 1 and 2;

3. Detecting, measuring, and controlling any possible radioactive releases into the environment;

4. Recognizing and controlling all potential hazards in the work environment; and

5. Observing workers involved in fighting the fire and treating and caring for anyone who became sick or injured during the fire at Browns Ferry.

Items 1 and 2 were handled by the Division of Power Production's Emergency Center; item 3 by the Division of Environmental Planning's Environs Emergency Staff (see memo, Oppold to Fox, dated April 15, 1975); item 4 by health physicists and industrial hygienists of the Division of Environmental Planning (memos from Oppold to Green, dated April 21, 1975, copy attached, and from Oppold to Thomas, with copy to you, dated April 8, 1975, provide respective findings); item 5 by the Browns Ferry nurses and Division of Medical Services physicians.

There was no evidence of any acute severe injuries or illnesses to any of the employees examined and followup medical evaluations revealed no remedial health effects. No radioactive materials were detected in the environment around Browns Ferry above those normally measured, and inplant worker radiation exposure was no more than normally expected. Several airborne toxic materials were searched for; only carbon monoxide in unit 1 reactor building was measured in significant quantities. Because of the carbon monoxide levels and the unknown makeup of the soot/ash dust, the industrial hygienists were directed to stay inplant until carbon monoxide levels dropped to a safe level and until, in their professional opinion, workers could safely enter areas affected by the fire without self-contained air supplies.

At about 2330 the CECC secured for the night. Two representatives of the Division of Power Production took charge of their Emergency Center. All CECC participants left with these two individuals telephone numbers where each could be reached during the night.

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PAGE 2 of 3

Harry S. Fox
April 28, 1975

SUMMARY OF CECC ACTIVITIES FOR MARCH 22-23, 1975

At approximately 0800 on March 23 I went to the CECC and remained until about 1615. The log is available on this time period if needed.

TVA's Radiological Emergency Plan and the activation of the CECC were effectively implemented in my opinion. In retrospect, the role of the CECC may have been minimal in this incident, but it certainly provided a necessary function.

James A. Oppold

JAO:DV
Attachments
CC (Attachments):
E. Floyd Thomas, 716 EB, Chattanooga (Attn: J. R. Calhoun)
R. L. Craig, 320 EB, Chattanooga

MC: E. A. Belvin, ROB, MS
Wayne Holley, 150 401B, C
P. A. Krenkel, 268 401B, C
R. S. Rainey, ROB, MS
D. M. Trayer, ROB, MS

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APPENDIX 7

FINAL REPORT OF TVA PRELIMINARY INVESTIGATION DATED MAY 7, 1975

FIRE AT BROWNS FERRY NUCLEAR PLANT

TENNESSEE VALLEY AUTHORITY

MARCH 22, 1975

FINAL REPORT

OF

PRELIMINARY INVESTIGATING COMMITTEE

MAY 7, 1975

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B. Key Photographs of Fire Area (Referenced on page 38)

I. INTRODUCTION

A preliminary investigating committee was established on March 23, 1975, to conduct an early fact-finding investigation of the fire and related events which occurred at the Browns Ferry Nuclear Plant on March 22, 1975. The interdivisional committee consists of the following members:

H. S. Fox, Chairman - Division of Power Production

Charles Bonine, Jr. - Division of Construction

Harry S. Collins, Reporter - Manager of Power's office

David G. Powell - Division of Law

M. N. Sprouse - Division of Engineering Design

Felix A. Szczepanski - Manager of Power's office

The committee's charter is included as appendix A. The committee reported to the plant on March 24, 1975, to initiate its investigation of the fire. A preliminary assessment of the damage was made, numerous interviews were conducted, and a preliminary report of the committee's findings was transmitted to the Manager of Power on April 7, 1975.

II. PURPOSE AND SCOPE

The purpose of this report is to present the committee's findings of facts on conditions and events relative to the fire and to provide a point of reference for other evaluations which may be required.

This report describes events leading up to, during, and after the fire until each unit was placed in the cold shutdown condition.

III. FINDINGS

A. Construction and Operational Status of Plant at the Time of the Fire

Units 1 and 2 were operating at normal full-load capacity, and construction work was proceeding on unit 3.

B. Relevant Design and Construction Features

1. Plant

A positive air pressure is maintained in the control bay, which includes the cable spreading room, with respect to the reactor building. In order to maintain the pressure differential, all penetrations between the control bay and reactor building are designed to provide an air pressure seal. A vertical cross section of the reactor building, control room, and spreading room, which is the area under consideration, is shown as figure 1.

2. Electrical Cable Penetrations

Electrical cable penetrations provide a means of routing cables through barriers such as floors and walls. They can be in the form of conduit or special fabricated steel sleeves.

a. Wall Penetration as Designed

The cable penetration where the fire started is contained in a 48-inch-square opening through the concrete wall separating the units 1 and 2 cable spreading room from the unit 1 reactor building. Division of Engineering Design (DED) drawings require the installation of a 1/2-inch-thick steel plate bulkhead slightly less than 48 inches square in the center of the opening in the concrete wall. Ten openings are cut in the bulkhead plate, and two stacks of five 18-1/2- by 5-1/2-inch steel sleeves are welded into the openings. The steel sleeves are 6 inches long and extend 3 inches on each side of the bulkhead centerline. The vertical clearance between the sleeves is 4 inches, and the horizontal clearance is 5 inches. The steel bulkhead assembly is framed and attached to the wall inside the concrete opening by 1-1/2- by 1-1/2- by 1/4-inch mounting angles. The cable trays abut the wall and are secured to angle iron extending horizontally across the face of the wall. Only the cables extend through the wall penetration. (See figures 2 and 3.)

The design requires that the penetration sleeves, with the cables installed, be filled with polyurethane foam to create an air pressure seal. (See figure 3.) A

flameproofing compound, Flamemastic 71A, was specified to be applied 1/8 to 1/4 inch thick over the foam and the cables on both sides of the bulkhead for a distance of 12 inches to form a fire stop.

Field tests were conducted on a typical cable penetration at the site in 1973. Later a test sample was sent to the TVA Singleton laboratory for fire performance testing. A DED staff engineer evaluated the test data and approved the results.

b. Wall Penetration as Originally Constructed

To facilitate sealing of the penetrations and to provide a practical starting point for filling the space around the cables with polyurethane foam, a means of forming a dam is required to prevent the liquid foam from flowing out of the sleeves. A preformed, resilient polyurethane foam was cut to size for insertion into the sleeve opening to form a dam. Other materials, such as styrofoam, were also used in some instances as a back dam. Pourable polyurethane foam was applied over and around the installed cables; after hardening of the pourable polyurethane foam, sprayable Froth Pak Insta-Foam polyurethane was used to finish filling the sleeve. The pourable foam is used since it more easily fills the voids between the cables. The sleeve and 12

inches of cables on both sides of the penetration were then coated with Flamemastic to provide the fire stop. The steel bulkhead as constructed was mounted in the opening with the centerline 3 inches from the surface of the wall on the reactor building side and 23 inches from the surface of the wall on the spreading room side, as indicated by dimensions on figure 4. Materials in addition to polyurethane foam were used to form the pressure seal.

3. Materials Used in Penetrations

Materials used for construction of fire stops, air pressure seals, and resealing after modifications to penetrations are described on table 1.

Diligent efforts are being made to secure from the manufacturers the physical and chemical properties of the materials in table 1, items 1-8, and will be made available if received.

For small leaks in cable penetrations, RTV silicone rubber was typically used as a sealant. For larger leaks, resilient polyurethane foam was typically used as a dam or a plug to contain the RTV silicone rubber or polyurethane foam.

4. Status of Penetration at Time of Fire

The penetration in which the fire started had been originally

sealed with polyurethane foam. There is evidence that the penetration had originally been coated with Flamemastic on the spreading room side. An examination after the fire indicates that Flamemastic had been applied to the unit 1 reactor building side of the penetration at some time prior to the fire and modifications which made resealing necessary.

Additional cables had been pulled through the penetration since initial installation. In order to make an opening for additional cables through the penetration, holes were punched with a wooden stick similar to a broom handle. This resulted in breaching any flameproofing that had been applied. This process usually resulted in pieces of polyurethane and Flamemastic in the penetration being knocked onto the cables on both sides of the penetration. This procedure has been generally followed when additional cables are pulled through completed penetrations. Fragments of these materials were observed on the cables in a number of other trays adjacent to the penetrations.

C. Activities Preceding the Fire

The areas within the plant are designed such that the air movement from one plant area to another will always be toward the area of possible higher radiation. This is controlled by supply and exhaust fans. The area of the reactor building and

refueling floor (secondary containment) is the area of lowest pressure, and any leakage between secondary containment and other plant areas will be inleakage into the secondary containment.

Under certain conditions, the standby gas-treatment system must exhaust air from the reactor building to maintain a negative pressure. In order not to exceed the capacity of the system, inleakage to the reactor building must be kept at a minimum.

In the completed plant, the refueling zone is common for all three reactor units. During construction an airtight partition is required between operating units and those under construction; and one exists between operating units 1 and 2, and unit 3 which is under construction. Before this partition between units 2 and 3 could be removed, it was necessary to ascertain the degree to which the standby gas-treatment system could handle the added inleakage from the unit 3 reactor building. The Division of Power Production (DPP) was requested to run leakage tests on the units 1 and 2 reactor buildings. The results of those tests indicated that leakage had to be reduced to a minimum if the unit 3 reactor building could be included and inleakage remain within the requirements of the units 1 and 2 technical specifications.

In a program to reduce leakage, the Division of Construction (DEC) wrote workplan 2892. The plan required (1) that all leaks be identified and listed, (2) that leaks be sealed, and (3) that work be verified and signed off by an engineer.

The method for detecting air leaks was largely left to the discretion of the engineer in charge. Several methods have been employed at Browns Ferry. These include smoke devices, soap solutions, and candles. The movement of the flame of a candle was an effective method in locating leaks in dimly lighted areas and generally was the method used.

A list was made of all leaking penetrations. These were identified by elevation and wall location, cable tray identification, and conduit number. The list was given to the electrical craft supervision with a requirement for the foreman to sign off for completed items.

Checking the resealed penetrations was basically the same as inspecting for leaks. However, experience had shown that as the number of leaks was reduced, the differential pressure increased; and other penetrations that originally did not seem to leak began to show airflow. Therefore, the inspectors (engineering aides) were instructed to check all penetrations in their assigned areas. The inspectors were accompanied by electricians who sealed any leaking penetrations as they were discovered. The inspectors often aided the electricians by checking penetrations as they were being sealed.

A successful leakage test and its documented approval were considered as evidence of the pressure seal's integrity.

For production efficiency, application of the Flamemastic did not immediately follow the sealing activities but was applied at intervals when sufficient numbers of seals were made ready.

On March 22, 1975, DEC workers were in the spreading room, sealing and leak-testing cable penetrations between the cable spreading room and the reactor building, when (at approximately 1220 hours--all times are Central Daylight Time) some of the sealant material in the penetration was unintentionally ignited at cable tray VE.

D. Fire

1. Spreading Room Area

a. Sequence of Events

Six men were working in the units 1 and 2 cable spreading room, checking conduit and cable penetrations for air leaks and sealing leaks.

An engineering aide and an electrician were checking cable penetrations through the wall between the spreading room and the unit 1 reactor building, in a window containing 10 cable trays in 2 vertical rows of 5 trays.

The engineering aide was using a candle flame to detect air leaks.

A differential air pressure existed between the spreading room and the reactor building, with the reactor building having a slightly negative pressure and thus causing air to flow from the spreading room through leaks into the reactor building.

The aide detected a strong air leak in the penetration for the second tray from the bottom on the west row.

The leak was caused when additional cables were pulled through the penetration, which resulted in breaching the originally installed air pressure seal and fire stop.

The electrician could not reach the penetration since it was recessed into the wall farther than he could reach.

The aide volunteered to seal the leak for the electrician. The electrician handed the aide two pieces (about 2 inches by 2 inches by 4 inches) of resilient polyurethane foam which the aide inserted into the hole.

After inserting the resilient polyurethane foam into the leak, the aide placed the candle about 1 inch from the resilient polyurethane foam.

The airflow through the leak pulled the candle flame into the resilient polyurethane foam, which sizzled and began to burn.

The aide immediately told the electrician that the candle had started a fire.

The electrician handed the aide a flashlight, which was used to try to beat out the fire with no success.

Another construction worker heard the aide state that there was a fire and gave the aide some rags to use to smother the fire, which was also unsuccessful.

The electrician called for fire extinguishers.

When the rags were pulled away from the penetration, they were smoldering.

Meanwhile, the other worker brought a CO_2 fire extinguisher to the aide.

The fire burned for about 1-1/2 minutes before the first extinguisher arrived.

The entire contents of this CO_2 extinguisher was emptied on the fire. The fire appeared to be out.

About 1/2 to 1 minute later, the fire started up again.

The aide stated that the fire was now on the reactor building side of the wall.

Two construction workers left the spreading room for the reactor building to fight the fire.

The electrician took two fire extinguishers to the aide who remained in the spreading room. Each extinguisher gave only one good puff.

When the aide received the third extinguisher, he heard a fire extinguisher being discharged on the reactor building side of the wall.

As the aide prepared to discharge the fourth extinguisher, the spreading room CO_2 system alarm was sounded; and all workers evacuated the spreading room.

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A plant operator, assistant shift engineer (ASE), after ensuring that no workers were in the spreading room, attempted to initiate the spreading room fixed CO_2 system from outside the west door to the room but was unable to do so because it had been deenergized while workmen were in the spreading room.

The ASE then ran to the east door of the spreading room, where he restored the electrical power and initiated the CO_2 system, which then operated properly.

Another ASE later operated the CO_2 system a second time.

After the CO_2 system had been operated the second time, the first ASE checked the spreading room and found that the fire had restarted.

He then directed the fire brigade in fighting the fire in the spreading room.

At 1310 hours, the ASE in charge of the reactor building fire requested the Athens Fire Department to come to the plant.

Employees from the Athens Fire Department assisted in fighting the spreading room fire.

The spreading room CO_2 system was operated one additional time.

An off-duty shift engineer (SE) arrived about 1500 hours and took charge of firefighting in the spreading room and relieved the ASE.

The spreading room fire was extinguished between 1600 hours and 1630 hours, primarily by using dry chemicals.

b. Description of Fire in the Spreading Room

The material ignited by the candle flame was resilient polyurethane foam.

Once the foam was ignited, the flame spread very rapidly.

After the first application of the CO_2, the fire had spread through to the reactor building side of the penetration.

Once ignited, the resilient polyurethane foam splattered as it burned.

After the second extinguisher was applied, there was a roaring sound from the fire and a blowtorch effect due to the airflow through the penetration.

The airflow through the penetration pulled the material from discharging fire extinguishers through the penetration into the reactor building.

Dry chemicals would extinguish flames, but the flame would start back up.

c. Equipment

Portable CO_2 and dry-chemical fire extinguishers were used in the spreading room fire.

The spreading room fixed CO_2 system was activated three times.

Breathing apparatus (air packs) received limited use in the spreading room.

The doors to the spreading room were kept open most of the time to assist in keeping smoke out of the control room.

An inplant fire hose was run from an outlet in the turbine building to the spreading room. This was not used.

The Athens Fire Department made available in the spreading room about 5 gallons of an agent which, when combined with water, forms "light water." This was not used.

Athens Fire Department employees discussed with the SE the possibility of using water on the fire in the spreading room.

No water was used in the spreading room since there was no assurance that the cables were deenergized.

d. Time of Events

(Approximate times shown with ∿)

∿ 1220	Fire started in penetration
∿ 1230	Two construction workers leave spreading room for reactor building
1235	Plant fire alarm sounded. Fire logged in SE's log
∿ 1237	First fire extinguisher discharged in reactor building
∿ 1240	CO_2 alarm sounded in spreading room; CO_2 system operated
?	Spreading room CO_2 system operated second time

?	ASE assumes direction of fire brigade in fighting fire
?	Spreading room CO_2 system operated third time
∿ 1500	SE assumes charge of spreading room firefighting
∿ 1600-1630	Spreading room fire extinguished

e. Reporting the Fire

Two construction workers left the spreading room at about 1230 hours to go to the reactor building to fight the fire.

One worker stopped at post 8D, a construction portal manned by the Public Safety Service (PSS), and informed the public safety officer on duty that there was a fire in reactor building number 1 and took the fire extinguisher with him to use in fighting the fire.

The officer immediately called the SE and reported a fire in unit 1 reactor building.

The ASE who received the fire report immediately gave the message to the SE and the unit 1 operator and then proceeded to the control room and switched the fire alarm to assure continuous sounding.

The unit operator (UO) immediately began to announce over the PA system that there was a fire in the unit 1 reactor building.

At this time, operators in the control room did not know the exact location of the fire.

An ASE located the fire in the unit 1 reactor building shortly after the construction workers had begun to fight it there. He telephoned the exact location to the operators in the control room.

Shortly thereafter another ASE in the reactor building reported the spreading room fire to the operators in the control room.

2. Reactor Building Area

a. Sequence of Events

When workers in the spreading room saw that the fire had spread into the reactor building, two construction workers left the spreading room and proceeded to the reactor building to fight the fire.

One worker told the public safety officer at post 8D that there was a fire in the reactor building and took a fire extinguisher with him. The other construction worker

proceeded to the reactor building where he met a third worker; each of the three workers took a fire extinguisher to the fire.

All three workers arrived at the fire at about the same time. It was burning in the trays which were 20 feet above the second floor of the reactor building. One moved a ladder, already at the scene, next to the fire. Another worker climbed the ladder and discharged a dry-chemical extinguisher on the fire. This application knocked down the flames, but the fire flared up again.

One of the workers alerted other workers on the second level of the unit 1 reactor building of the fire.

The worker who applied the first extinguisher was affected by the smoke and fumes around the cable trays at the top of the ladder.

The unit 1 control room operator was informed by telephone of the precise location of the fire by a plant operator on the scene.

An ASE then arrived and, along with another operator, discharged a CO_2 and a dry-chemical extinguisher

simultaneously on the fire. The ASE assumed charge of firefighting activities. Construction workers were instructed to leave the operating units.

Smoke was becoming so dense that breathing apparatus was required; approximately 5 minutes after it was requested, it was available. Until it arrived, CO_2 was applied to the cable trays from the floor.

After the breathing apparatus (air packs) arrived, it was utilized in fighting the fire until visibility became so bad that the workers could not get near the fire. The smoke backed them up to the area of the reactor building closed cooling water system heat exchangers.

The ASE left the fire to assist in unit shutdown. An assistant unit operator (AUO) assumed charge of firefighting activities. The first floor of the reactor building was also evacuated. The AUO went to the control room due to some ill effects of the smoke. Another ASE assumed charge of firefighting activities.

Power to the elevator was lost. The second floor of the reactor building was then evacuated. Some time

was utilized to check 5 floors of the reactor building for the elevator to ensure that no one was trapped on the elevator. A head count was made, and from that point on a count was kept of all personnel leaving and entering the reactor building.

About 1330 hours, lighting was lost in the reactor building.

Limited firefighting was resumed in the reactor building for a period between 1430 hours and 1500 hours. A wire was used to rig a guideline. At this time the fire was still confined to the area in the cable trays near the north wall and had not proceeded very far on the south trays.

At this time, the doors between units 1 and 2 were opened, which improved visibility on the second level of unit 1 to about 5 feet.

At about 1630 hours, the SE who had been directing activities in the spreading room took charge of fire-fighting in the reactor building in order to concentrate activities there. The SE consulted the plant superintendent frequently during fighting of the reactor building fire.

On inspection of the fire at 1630 hours, the major fire was in the cable trays running south from the penetration, with a smaller fire in the cable trays running west from the penetration.

The SE established a routine of sending 2 to 3 people in at a time to fight the fire, using dry chemicals primarily.

Shortly after 1630 hours, temporary d.c. lighting was strung on the second level of unit 1.

A rope was utilized as a guideline, which assisted employees from the Athens Fire Department in approaching the fire to inspect it. The SE went into the vicinity of the fire between 1730 hours and 1800 hours.

On one of his trips into the second level, the SE laid out the fire hose installed there and checked to ensure that water was available. The plant superintendent authorized the use of water as an emergency backup, for example, in case a worker's clothing caught fire. Otherwise, there was a decision not to use water on the fire due to the electrical shock hazard. The Athens fire chief suggested that water would be the best thing to use on the fire if it could be used.

The SE suggested to the plant superintendent that water be used on the fire. The superintendent made the decision to allow the Athens Fire Department employees to use water on the fire.

Water was initially applied to the trays running west; however, from the floor level, the water would effectively reach only the bottom tray. Athens Fire Department employees attempted to utilize one of their nozzles on the hose, but the thread did not match; and the nozzle came off when pressure was applied.

Water was also applied to the fire in the cable trays along the north wall and successfully extinguished it.

Firefighters began using Chemox respirators as the supply of compressed air for the air packs ran low.

The SE and two other operations workers entered the area of the fire to utilize water to fight the fire. The SE took the hose and climbed within four feet of the fire with assistance of the other two men. He sprayed water on the fire in the south cable trays for approximately 10 seconds, which extinguished the fire.

The fire hose was left stuck in a position so that it continued to apply water to the south cable trays.

The second level was entered again and water reapplied. It was then determined that the fire was out. There were subsequently some reports of sparks, but investigation failed to reveal any further fire.

During the course of the fire, it was noticed that a small diameter station control air line under about 90 pounds of pressure, running along the north wall, had parted. The line was later isolated.

Several fire extinguishers were discharged early in the the fire from the third floor through an opening in the floor, but all missed the fire in the cable trays since the opening was not directly over the fire.

b. <u>Description of Fire in Reactor Building</u>

The fire was initially observed in the lower cable trays, extending out from the penetration a distance of 2 to 4 feet. Height of the flames varied from a few inches to a few feet, dying down as extinguishing materials were applied and flaring up between applications. The flames were coming straight up.

Some polyurethane foam was flowing from the penetrations into the trays, and bright yellow flames were coming from the penetrations.

The fire did not advance significantly into the south trays until after 1500 hours.

Scaffold boards had been previously placed below the trays in the unit 1 reactor building, near the cable tray penetration where the fire started. These boards were used to work from in pulling cables through the penetration. These boards were charred by the fire. The charring did not extend to the side away from the fire, indicating little influence as fuel for the fire.

c. Equipment

Portable CO_2 and dry-chemical fire extinguishers were used in the reactor building fire.

MSA air packs were used that had a rating of 30 minutes for moderately heavy activity of the user. A cascade system of large air cylinders was available for charging the packs, but the supply was eventually depleted. There are no air compressor facilities at the plant to fully recharge the air packs. The charges in some

air packs did not last 30 minutes. Air packs from Athens Fire Department were also used along with their recharging facilities on their truck and at their station in Athens.

MSA Chemox respirators were used. Several users experienced difficulty when using these for very strenuous activity.

The fire hose and nozzle provided in the second level of the reactor building functioned properly and successfully extinguished the fire.

A nozzle from the Athens fire truck did not fit the threads on the hose on the second floor of the reactor building.

Ladders present on the second level of the reactor building were utilized.

Temporary d.c. lighting was utilized.

A wire and a rope were utilized as guidelines.

A fire hose was laid out on the third floor of the reactor building but was not utilized.

d. Time of Events

∿ 1230	Two construction workers leave spreading room for reactor building
∿ 1237	First fire extinguisher discharged in reactor building
∿ 1240	Unit operator informed of exact location of fire in reactor building
?	Air packs requested and received
∿ 1310	ASE requested that Athens Fire Department come to the plant
∿ 1330	Lighting lost in reactor building
∿ 1645	Temporary d.c. lighting installed
∿ 1835	Water applied to fire
∿ 1930	Fire determined extinguished

e. Minor Fires on Thrusday, March 20

There were two minor fires on Thursday, March 20, arising from the use of candles for leak-testing in electrical cable penetrations different from the penetration involved in the March 22, 1975, fire. In the first fire, the candle flame ignited some RTV silicone rubber. The construction worker using the candle extinguished the flareup with his fingers.

In the second fire, the candle flame ignited dust and debris in the cable tray. The fire lasted about 30 seconds

and was extinguished with a discharge from a CO_2 extinguisher.

The first fire was reported orally to construction supervisory workers; the second fire was entered in the SE's log and reported in writing to construction supervisory workers.

There was no damage from either fire.

E. Effect on Plant Systems and Operations

1. Status of Plant Operations Prior to Fire

At the time of the fire on March 22, 1975, units 1 and 2 were each producing approximately 1,100 MWe gross. Unit 1 was declared in commercial operation on August 1, 1974, and unit 2 on March 1, 1975.

2. Unit 1

The ignition of the fire in the cable penetration has been established as accurately as possible to have occurred at 1220 hours on March 22, 1975. The first indication of its effect on unit 1 operation came 20 minutes later, at 1240 hours. This was 5 minutes after the UO's were notified of the fire and the alarm initiated at 1235 hours.

The first effect on the unit was almost simultaneous annunciation of several events: residual heat removal (RHR)

or core spray (CS) automatic blowdown permissive, reactor water level low-automatic blowdown permissive, and core cooling system/diesel initiate.

At this point the UO observed that normal conditions of reactor water level, reactor steam pressure, and drywell atmosphere pressure existed.

Over the next 7 to 8 minutes, a mounting number of events occurred, including the automatic starting of RHR and CS pumps, high-pressure coolant-injection (HPCI) pump, and reactor core isolation coolant (RCIC) pump; control board indicating lights were randomly glowing brightly, dimming, and going out; numerous alarms occurring; and smoke coming from beneath panel 9-3, which is the control panel for emergency core cooling systems (ECCS). The operator shut down equipment that he determined was not needed, such as the RHR and CS pumps, only to have them restart again.

When the reactor power became affected by an unexplained runback of the reactor recirculating pumps, the SE instructed the operator to reduce recirculating pump loading and scram the reactor. While this was being done, the recirculating pumps tripped off. The reactor was scrammed by the operator at 1251 hours.

The turbogenerator was then removed from service; steam from the reactor was bypassed around the turbine to use the condenser as a heat sink; and unneeded condensate, condensate booster, and reactor feed pumps were removed from service. One of each pump was left running to maintain reactor water level. Beginning at approximately 1255 hours and continuing for about 5 minutes, several electrical boards were lost, supplying control voltages and power voltages of 120, 480, and 4,160 volts a.c. and 250 volts d.c. These mainly affected reactor shutdown equipment.

As a result of the loss of these electrical boards and previous effects, many of the systems used in cooling the reactor after it is shut down became inoperative. This included the RHR system, core spray system, HPCI, and RCIC. This is attributed to loss of valve control signals, valve power voltage, motor control signals, motor power voltage, or a combination of these. In addition, many of the instruments and indicating lights were put out of order. Also, the outboard main-steam isolation valves (MSIV's) closed. This isolated the steam generated by reactor decay heat from the condenser heat sink. The valve closure also isolated the steam supply to the turbine-driven reactor feed pumps, and consequently this high-pressure source of water to the reactor was lost. At this

time the water input to the reactor was limited to the control rod drive pumps as a high-pressure water source since the steam pressure built to a pressure of 1,080 psi and was being relieved by automatic operation of the relief valves to the suppression pool.

Alternative systems were available and were used effectively to shut down and cool the reactor. This was accomplished by manual opening of the relief valves to reduce reactor pressure below 350 psi where the condensate booster pumps could pump an adequate supply of water to the reactor. The reactor water level decreased during this operation, but it did not drop below a point 48 inches above the top of the active fuel and was returned to normal level by 1345 hours.

Early in the chain of events, the diesel generators started and were allowed to run on standby. During a short period of time the four diesel generators were used to supply their respective shutdown buses. About 1443 hours one of the diesel generators became unavailable.

Soon after the loss of electrical boards, operating workers began attempts to restore the electrical supplies.

Initially, this was generally unsuccessful. Attempts to manually position valves and locally operate the equipment were hampered by darkness and the smoke and fumes from the fire filling the reactor building, requiring the use of air-breathing packs. Some smoke and CO_2 came into the units 1 and 2 control room from firefighting efforts in the spreading room, but it was not necessary to vacate the control room at any time. Two of the operators in the unit 1 control area donned breathing apparatus for a short period of time because of the smoke and fumes. To establish the electrical supply boards, maintenance electricians joined the operators in isolating faulted circuits in order that the boards could be reenergized. This was done over several hours, and needed equipment to provide suppression pool cooling and reactor long-term shutdown cooling was gradually made available.

With adequate electrical power, along with some manual valve alignment, the operators established suppression pool cooling at 0130 hours on March 23, 1975, 12 hours 39 minutes after the unit 1 reactor was scrammed. Normal reactor shutdown cooling was achieved at 0410 hours on March 23, 1975, 15 hours 19 minutes after the unit was scrammed.

3. Unit 2

Nine minutes after unit 1 was scrammed, abnormal events began occurring on unit 2. At 1300 hours the 4-kV shutdown bus 2 deenergized; and the operator observed decreasing reactor power, many scram alarms, and the loss of some indicating lights. The operator put the reactor in shutdown mode, and it scrammed at 1300 hours.

The turbine was immediately tripped, along with the reactor feed pumps. In approximately 4 minutes after scram, the MSIV's closed, isolating the reactor steam from the condenser heat sink and the reactor feed pumps steam supply. RCIC was immediately initiated for reactor water level control and the HPCI to aid as a heat sink for the steam being generated in the reactor by decay heat. These two systems tripped several times over the next hour, and at approximately 1345 hours HPCI became unavailable. RCIC continued to run and supply high-pressure water to the reactor.

When suppression pool temperature began to increase from relief valve steam heating, RHR suppression pool cooling was established at 1320 hours; and the temperature of the water in the torus did not exceed 135° F.

When the MSIV's closed, reactor pressure was relieved by manual operation of the relief valves. Manual operation of the relief valves was lost at 1320 hours and the relief valves lifted intermittently on pressure until 1415 hours, when manual operation was restored; and the reactor was depressurized by use of the relief valves.

At 2010 hours the MSIV's were reopened, making the condenser heat sink available. At 2020 hours on March 22, 1975, equipment was made available to establish operation of the RHR system to be used for reactor long-term shutdown cooling. This was 7 hours 20 minutes after the unit was scrammed.

4. Detailed Operating Events, Operator Action, and Equipment Response and Nonresponse

Tables 6 and 7 provide the sequence of events, operator action, and equipment response which occurred during the fire and until conditions were stabilized (initiation of shutdown cooling) on both units 1 and 2. The events listed on tables 6 and 7 are arranged chronologically, with the best possible establishment of times without the benefit of complete operator logs.

Most of the time, particularly during the early stages of the fire, operators were too busy to log the frequent events and actions. Some of the times and facts were established by charts and printers but for the most part by interviews with operating personnel, both individually and in groups.

5. Status of Major Plant Equipment and Systems and Plant Parameters at the Initiation of Reactor Long-Term Shutdown Cooling

a. Unit 1 at 0410 hours on March 23, 1975

Reactor coolant temperature 360° F

Reactor vessel water level normal

Suppression pool water level +5"

Suppression pool water temperature 153° F

Control rod drive pump and condensate pumps providing makeup water to reactor vessel

Standby liquid control system available

Core neutron monitoring provided by two temporary source range monitors connected outside primary containment with the monitors manned by a licensed reactor operator in communication with a licensed reactor operator in the control room

Primary and secondary containment integrity being maintained

All 4-kV shutdown boards available

Shutdown bus 2 available and supplying offsite power to the shutdown boards

Remote indications (amps, watts, and volts) being read locally at shutdown boards where equipment operation required

Diesel generators A, B, and D available and operable from shutdown boards—diesel generator C unavailable because of control cable problems

RHR loop I pumps and valves available

RHR loop II pump B and valves available

Control for 3 RHR pumps available from control room; control from local stations for most valves

All loop I and loop II core spray pumps and valves available

Four relief valves remotely operable from unit control board

No automatic initiation of diesel generators, core spray system, or RHR system in low-pressure coolant-injection (LPCI) mode available

Suppression pool cooling in service

Suppression pool water level indication and drywell pressure indication operable

Train A of standby gas-treatment system operable

Control rod drive pump in operation--system flow and pressure indication unavailable

Process computer in service (40 analog inputs damaged by fire)

Telephone communication out of service for unit 1 reactor building, offgas vent building, and stack; in service for other areas

Liquid monitor on the effluent from the reactor building closed cooling water system, raw cooling water, and residual heat-removal heat exchangers out of service.

Grab samples of effluent water taken periodically by chemical laboratory personnel.

b. Unit 2 at 2240 hours on March 22, 1975

Reactor coolant temperature 260° F

Reactor vessel water level normal

Control rod drive and condensate pumps providing makeup water to reactor vessel

All RHR pumps operable

HPCI pump inoperable

Core spray loop I pumps A and C and RHR loop I pumps A and C operable only from shutdown boards

Conditions of long-term reactor shutdown cooling were considered normal

F. Damage Assessment (Cable Tray System, Conduit and Grounding System, and all Cables Routed Through These Raceway Systems)

This section summarizes the extent of the physical damage to the cables and the raceway systems involved in the fire at Browns Ferry on March 22, 1975, and indicates the detail to be found in a complete report provided by DED for use in the restoration program. The complete report is numbered BF-DED(BHP-1).

Excluded from the damage assessment are the effects of faults in these cables to mechanical and electrical systems; damage to other equipment resulting from products of combustion and the chemicals and water used in extinguishing the fire; possible structural and concrete damage; and damage outside the zone of influence of the fire. These areas are being evaluated in detail by others within TVA.

A fire consultant has been retained by DED to perform a thorough investigation with the purpose of providing a factually accurate and professional determination or assessment of the mechanisms and their interactions responsible for the initiation, propagation, magnitude, duration, and extent of damage of the fire. The consultant's report has not been received at the time of issuance of this report.

1. Zone of Influence of the Fire

It has been determined that the fire started when an open flame came into contact with material used as the seal around the cables where they penetrate the wall between the units 1 and 2 control bay spreading room and the unit 1 reactor building. Figures 5 and 6 indicate the area being considered in this description. Figure 6 shows the zone of influence of the fire. Figure 7 shows a cross section of trays near the point where the fire started. The cables and raceways in the spreading room were damaged approximately 5 feet north of the wall penetration; and the fire propagated along all trays, as marked on figure 8, in the reactor building on floor elevation 593. Many photographs were taken, and 10 key ones are included in this report as appendix B. Figure 9 shows affected trays and their intersections in single-line representation. Checkpoints used for routing cables on each cable tray are also shown. (See table 2 for loading of cable types onto each tray at each checkpoint.)

Visible damage in the reactor building was observed east along the double stack of 3 trays to the wall between units 1 and 2, south along the 4 trays to a fire stop approximately 28 feet from the wall between the reactor building and the control bay, and west along the double stack of 5 trays, for a distance of approximately 38 feet from the wall between units 1 and 2. Cables were also damaged on 2 of the 4 vertical trays from the top about 10 feet down, and cables in 1 of the other 2 trays were damaged about 4 feet down. Figures 10-12 show the zone of influence of the fire for all damaged or assumed-damaged conduits and grounding systems.

2. <u>Identification of Damaged Conduits, Cable Trays, and Cables Routed Through Raceways</u>

A total of 117 conduits and 26 cable trays was damaged by the fire, and it is assumed that all supports for the raceway system were also damaged. There was a total of 1,611 damaged cables, and these are tabulated on 204 cable tablulation sheets prepared by DED. Table 3 is a sample sheet of the 204 cable tabulation sheets which show the purpose of each cable and other pertinent information needed by DEC to be used in a procedure for identification and removal of damaged cables.

This procedure is being written by DEC to require that the damaged portion of each cable be identified and measured

during its removal. This procedure will also require that a section of the undamaged portion of each cable be removed, identified, and stored for future reference. This section will be cut to assure that all manufacturer's data stamped on the outer jacket will be included in the sample.

As of this date there have been 1,169 cables identified as damaged for unit 1, 75 for unit 2, 27 for unit 3, and 340 common to plant. Of the total cables identified and listed in table 3, it was determined that a total of 628 safety-related cables was damaged. These are grouped into categories shown on table 4.

The bare ground cable used for grounding the cable tray system was also damaged by the fire. It was routed along the 480-volt power trays FM, FK, and FO-ESII through the zone of influence.

3. Materials Available as Possible Fuel For the Fire

Of the 1,611 cables, there were 65 different-type cables involved in the fire, as listed on table 5. Figure 7 shows a cross section of the cable trays where the fire started. (See table 2, sheets 8 and 9, for the type cables found there.) These types are representative of each voltage level tray in the area. Types WBB through WNF are power and control cables manufactured in accordance with TVA standard specification and are composed of

insulating material footnoted on table 5, sheets, 2, 3, and 4. The remaining types are signal cables which are specified and documented on numerous individual contracts. These are composed of insulating material footnoted in table 5, sheets 2, 3, and 4. In all cases, the actual types used will be verified in the removal of cables and will be included in the final DED report BF-DED(BHP-1). The filler materials in these cables and cable ties are included in the listing at the conclusion of this section.

Another possible "fuel" was the wall penetration pressure seal materials used between the spreading room and the reactor building. A typical penetration is shown in figures 2 and 3. The sealant material was polyurethane expandable foam, a pressure seal, which is covered with Flamemastic, a flameproofing compound. Another sealant material which is a possible fuel source would be the RTV silicone rubber compound used in sealing conduits through walls and in some cases to seal around new cables added through penetrations.

4. <u>General List of Materials Associated With the Fire</u>

a. Candle

b. Polyurethane foam, Froth Pak Insta-Foam

c. Polyurethane, pourable type

d. Polyethylene

e. Nylon

f. Cross-linked polyethylene

g. Polyvinyl-chloride

h. Mylar

i. Aluminum foil and rigid aluminum conduit

j. Polyolefins

k. Chlorosulfonated polyethylene

l. Neoprene

m. Fiberglass

n. RTV silicone rubber

o. Galvanizing material on raceways

p. Carbon

q. Thermoplastic nonhygroscopic cable filler material

r. Preformed, resilient polyurethane foam

s. Marinite panels

t. Styrofoam

u. Copper

v. Steel

w. Flamemastic 71A

G. Radiological Assessment

Based on interviews with the plant health physics supervisor and the plant chemical engineer, and information provided by the Plant Results Section and the Division of Environmental Planning, the following has been established.

1. Releases Within the Plant and Personnel Exposures

a. At the time of the fire, one health physics technician was present at the facility. As requested, off-shift technicians reported to the plant, with the health physics supervisor arriving at approximately 1600 hours. At one time as many as 9 health physics workers were onsite.

b. Direct radiation surveys conducted within the reactor building indicated there was no increase in direct radiation above normal levels.

c. Numerous samples to detect airborne radioactivity present within the reactor buildings showed that the only significant particulate or halogen isotope present was the isotope Rubidium 88, a daughter product of the fission gas Krypton 88, with a half-life of 17 minutes. The buildup of Rubidium 88 is attributed to the shutdowns of the reactor building ventilation systems during the fire.

d. Analyses of the samples showed the maximum concentration of this isotope approximated only 35 percent of the maximum concentration permitted under NRC regulations in 10CFR20 for a 40-hour workweek.

e. Following the fire, a number of individuals, including operations and construction workers, who were considered the most likely to have received internal radiation exposure from being in the unit 1 reactor building, were whole-body counted (on March 24 and 25). All whole-body counts showed no indication of internal deposition of radioactive material.

f. Based on dosimetry information, no plant individual is shown to have exceeded the daily radiation exposure limit; and the film badge readings for the Athens Fire Department employees indicated they received no detectable radiation exposure.

2. Releases From the Plant

a. As a result of the fire, the radiation detectors that monitor the ventilation air exhausted from the unit 1 and the unit 2 reactor buildings were made inoperable. The unit 2 monitor was restored at about 1900 hours on March 22, 1975, and the unit 1 monitor restored at 1600 hours on March 23, 1975.

b. During the course of the fire and the time the monitors were out of service, grab samples were taken from the units 1 and 2 exhausts on the reactor building roof starting at approximately 1645 hours and each hour

thereafter and analyzed in the plant radiochemistry laboratory to determine concentrations of radioactivity. Charcoal filter and particulate filter samples were also taken from these airstreams periodically during the event.

c. All other required building ventilation duct monitors and the plant stack release monitors remained operable.

d. Gamma spectrum analysis of the grab samples indicated that the principal isotopes present were Xenon 133, Xenon 133m, Krypton 85m, and the Rubidium 88 detected in the inplant air samples. Analysis of the charcoal samples indicated no detectable amount of iodine.

e. Review of the airborne release rate information shows that the total plant release rate was the highest at 2200 hours on March 22 and corresponds to about 8 percent of the technical specification allowable limit for gross activity release.

f. Liquid radwaste is discharged from the plant periodically and on a batch basis. The last batch released before the fire occurred was on March 19. While as a direct result of the fire the liquid radwaste monitor became

inoperable, no release from the plant was being made at the time; and the monitor was returned to operation on March 24 before the next batch was released.

3. Environmental Consequences

a. While not required, the Environs Radiological Emergency Plan was activated for precautionary purposes at approximately 1500 hours on March 22, with the Environs Emergency Staff remaining active until approximately 0500 hours on March 23.

b. A report on the radiological environmental consequences of the fire, made at the committee's request, is summarized below:

(1) Analyses of air particulate and charcoal filter samples collected by monitoring teams in the downwind direction from the plant, based on continual evaluation of data from the plant's meteorological station, show that no radioactivity except that due to naturally occurring radionuclides was detected in the environment.

(5) The report states that "Based on actual measurements and collected data, calculations show that during the incident at the Browns Ferry Nuclear Plant, amounts of radionuclides released to the environment were well below the plant technical specification limits. Conservative calculations show that the radioactivity released to the environment had a very minimal and insignificant environmental impact."

H. Personnel Injuries

Information provided by the TVA medical director states that 7 TVA employees (6 from DPP and 1 from DEC) reported to the Browns Ferry construction project medical office and the health station with complaints associated with smoke inhalation. Under the direction of a TVA physician, each was evaluated and treated by the nurses on duty and released with instructions to report immediately any delayed effects. Shortly after being seen, one of the employees reported the onset of generalized chest discomfort on respiration. He was referred immediately to a local hospital, where he was examined and released by the physician. None of the employees revealed evidence of severe effects from their exposure.

Followup medical evaluations revealed no residual effects from the activities and exposures associated with fighting the fire.

There has been no medical indication for lost time from work. Each employee was medically approved to resume full duties on the next scheduled work shift.

I. Administrative Controls

1. DPP-DEC Interface for Work by Construction Forces in an Operating Unit

a. Under DEC Quality Control Procedure BF-104, Administrative Procedures to Maintain Physical Separation Between Construction and Operating Units and Control of Work in Restricted Access Areas, all modifications and completion work required on a licensed unit by construction employees are done under a workplan. This procedure also specifies (1) that workplans can be written by either DEC or DPP, (2) must be approved by the DEC coordinator, and (3) the DPP coordinator will determine the level of review required within DPP and finalize approval with his signature.

b. BFNP Standard Practice BFA-28, Plant Modifications, describes how modifications to the plant will be requested, performed, and documented, including the approvals necessary, depending on whether the modification is categorized as safety related or nonsafety related.

c. The work being performed at the time the fire started was approved by the DEC coordinator and authorized by the DPP plant modification coordinator under BFNP workplan 2892 which was issued under BF-104 on March 7, 1975.

d. On workplan 2892, the work to be performed is described as follows: "Check electrical and mechanical sealing for secondary containment. (1) make a punch list of sleeves and cable penetrations that require sealing, (2) complete sealing, (3) verify and sign off areas that were found leaking."

A list of identified secondary containment air leaks is attached to the workplan.

e. The space provided for identification of drawings associated with the work has the letters N/A (not applicable) entered.

f. A review of workplan 2892 and applicable administrative procedures indicates the work being performed under this workplan was not processed as a modification under BFA-28 but was processed under BF-104 which does not require that an unreviewed safety question determination be made according to the provisions of 10CFR50.59.

2. Construction Work Control

With regard to the control of the work being performed by construction forces, the committee established the following:

a. There were no written procedures or work instructions covering the sealing and testing of penetrations for the original installation or the modifications except for notations on DED drawings.

b. At the time the fire started, the engineering aide whose assigned responsibility was to inspect the work (i.e., to find the air leaks) was actually doing the work himself (i.e., plugging the leaks) instead of the journeyman electrician.

3. Fire Reporting

a. The existence of a fire was not reported immediately by construction workers discovering the fire. When reported to the PSS officer manning construction portal post 8D, the exact location of the fire was not specified.

b. BFNP Standard Practice BFS3, Fire Protection and Prevention, instructs DPP personnel discovering a fire, whether in a construction area or an area for which DPP is responsible, to report the fire to the construction fire department, telephone 235. BFNP Fire, Explosion, and Natural Disaster

Plan instructs personnel discovering a fire to dial 299 (PAX). The construction extension cannot be dialed from the PAX system, and the plant extension cannot be dialed from the construction phone system.

c. Dialing instructions for reporting fires are located on telephones and are also included on the emergency procedure sheet posted at various locations in the operating areas.

4. Work Hazards Control

While control requirements exist for certain potentially hazardous work, e.g., welding and burning operations, no written procedures or instructions have been issued at Browns Ferry regarding the introduction into and use of potentially hazardous materials or substances in connection with construction work in operating plant areas such as ignition sources and flammables.

J. Other Findings

The possibility of sabotage was investigated, and no reason to suspect sabotage was found.

IV. OTHER GENERAL INFORMATION

A. Central Emergency Control Center (CECC)

1. The CECC was activated on March 22, 1975, during the Browns Ferry fire as a precautionary measure, although no

radiological emergency existed. The CECC was directed from the Edney Building in Chattanooga, beginning at 1525 hours CDT on March 22, 1975, by the Assistant to the Director of Environmental Planning. Other available members of the CECC were notified of the fire.

2. The CECC performed a valuable function--keeping the Nuclear Regulatory Commission (in Atlanta), the Alabama State Department of Public Health, and the Tennessee State Department of Public Health informed rather than fulfilling a requirement of the Radiological Emergency Plan (REP). The CECC was in direct communication with the DPP Emergency Control Center.

3. The CECC office was secured at 2230 hours on March 22, 1975.

B. <u>DPP Emergency Control Center</u>

1. The DPP Emergency Control Center in Chattanooga was established at 1510 hours on March 22, 1975, with the Chief, Nuclear Generation Branch, in charge. By 1630 hours, approximately 20 DPP staff members had assembled at the control center, including the division director and other key management personnel. The branch chief and others were in frequent communication with the superintendent at Browns Ferry. This management team participated in all major decisions associated with the plant operation and firefighting activities.

2. The major group of the staff assembled left at 2200 hours on March 22, 1975. A small group manned the DPP Emergency Control Center until 1500 hours on March 23, 1975.

C. Other Programs for Repair and Return to Service of Equipment

A number of programs have been initiated to evaluate various aspects of the fire and its consequence and return to service of the equipment. A memorandum from E. F. Thomas to R. H. Dunham and H. H. Mull dated March 28, 1975, subject "Repair of Damage Caused by the Cable Fire and Return to Service of Browns Ferry Nuclear Plant Units 1 and 2" has been issued and is being updated to provide directions for these efforts.

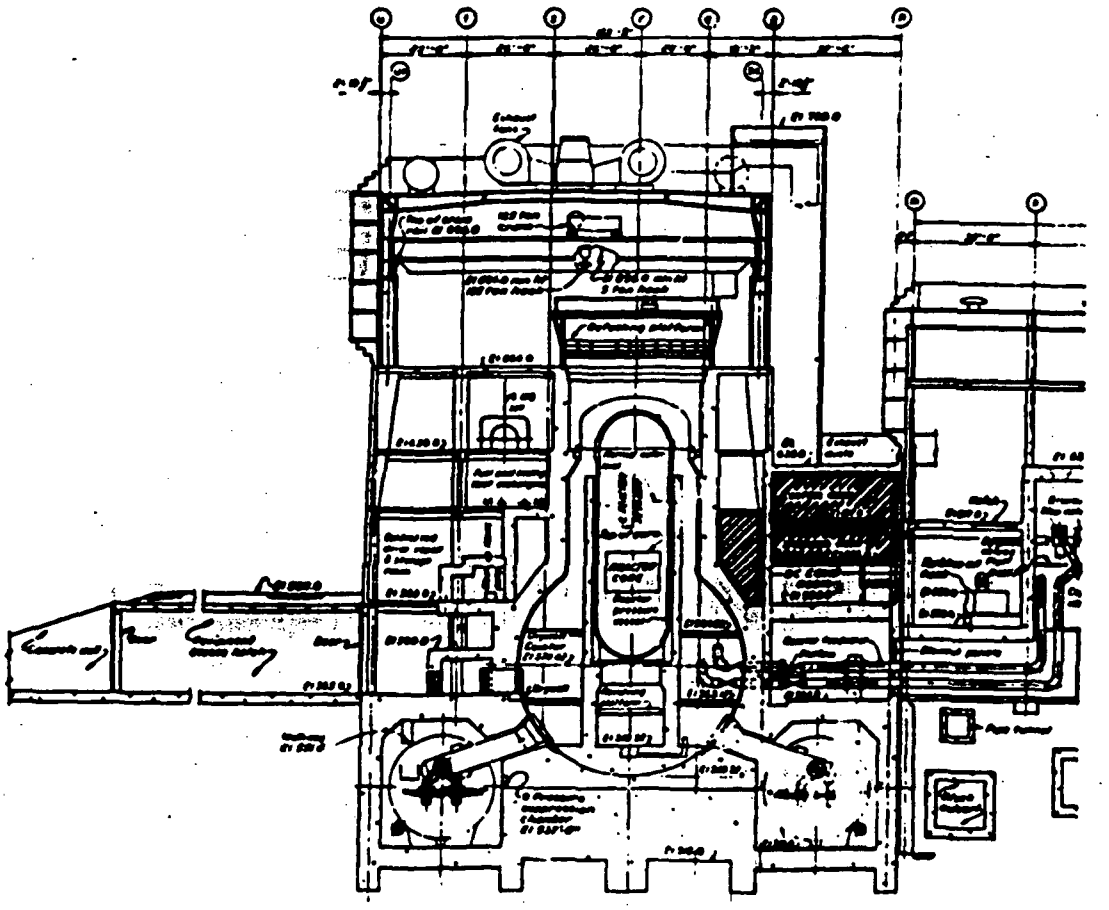

FIGURE 1

Vertical Cross Section

Reactor Building, Control Room,
and Spreading Room

FIG. 1

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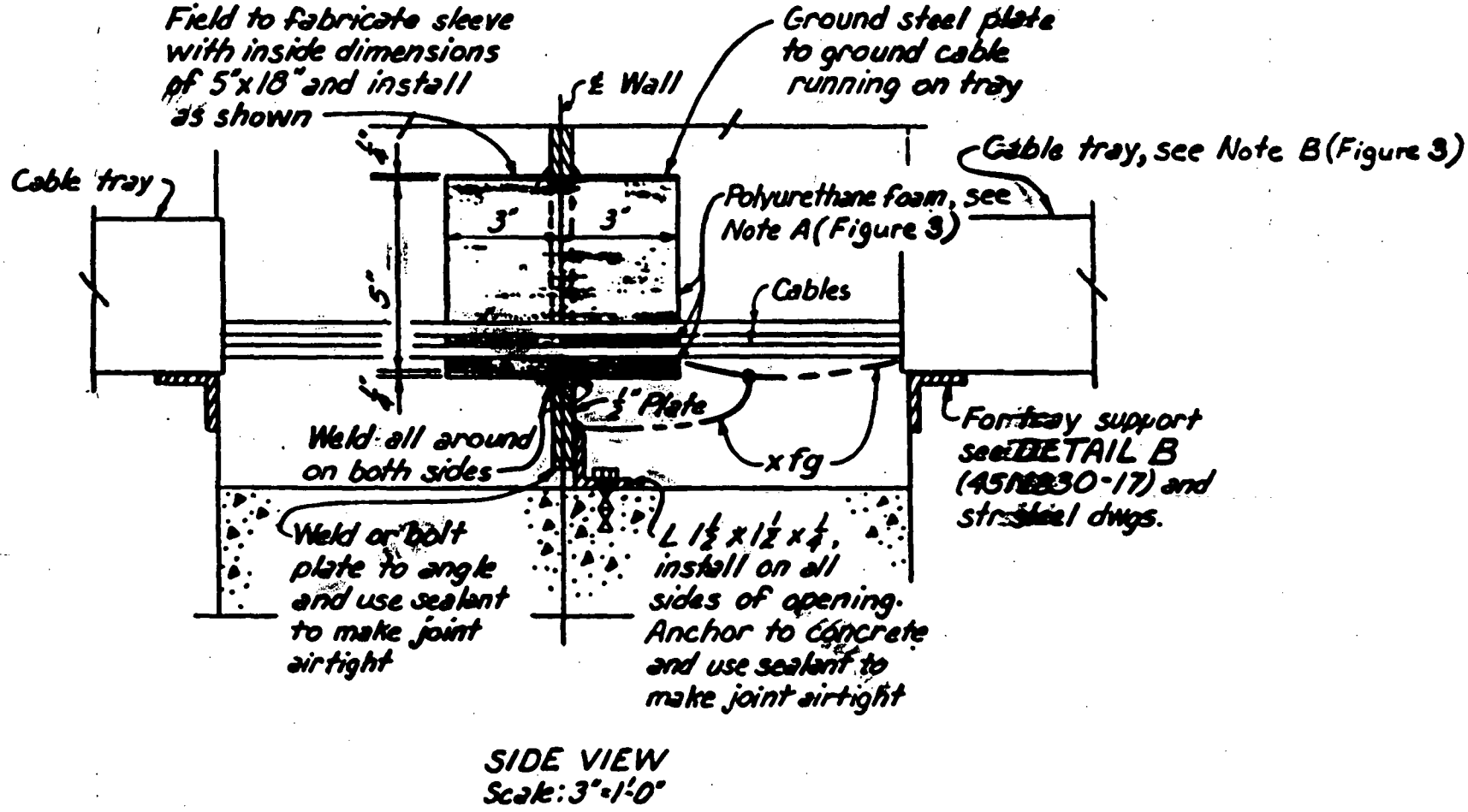
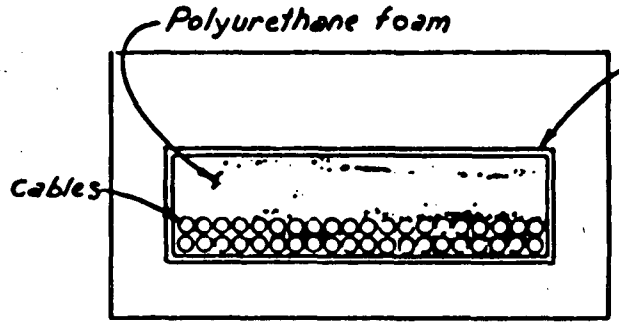
FIGURE 2
TYPICAL WALL PENETRATION



SIDE VIEW
Scale: 3"=1'-0"

FIG. 2



NOTES:

A. When all cables have been installed through cable sleeve, seal the remaining opening and voids with 6" ± of Urethane foam or equal before applying flame proofing compound, Flamemastic 71A or equal.

B. Apply approximately 1/8" to 1/4" of flame proofing compound on the steel sleeve and on both top and bottom of the tray and cables for 12" on both sides of the barrier.

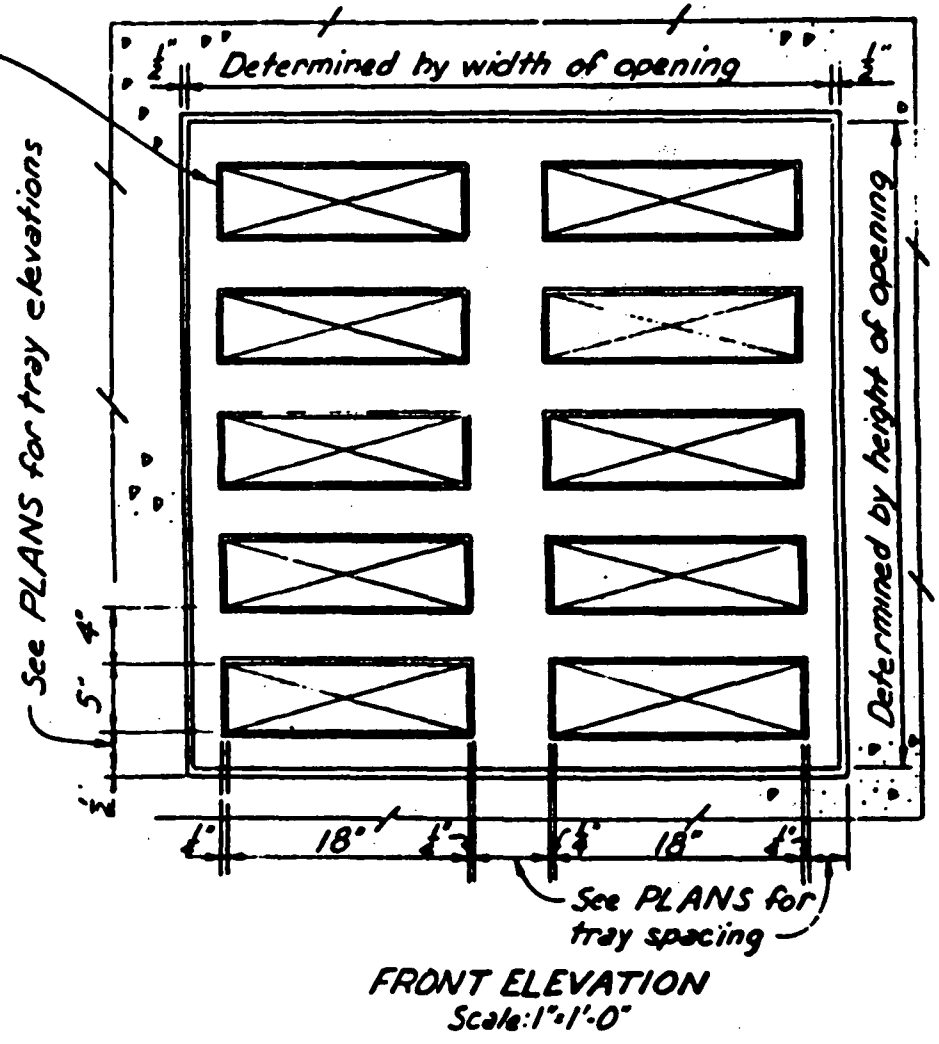


FIGURE 3
TYPICAL WALL PENETRATION

NOTE: FIRE STARTED IN SECOND PENETRATION FROM BOTTOM-TRAY VE

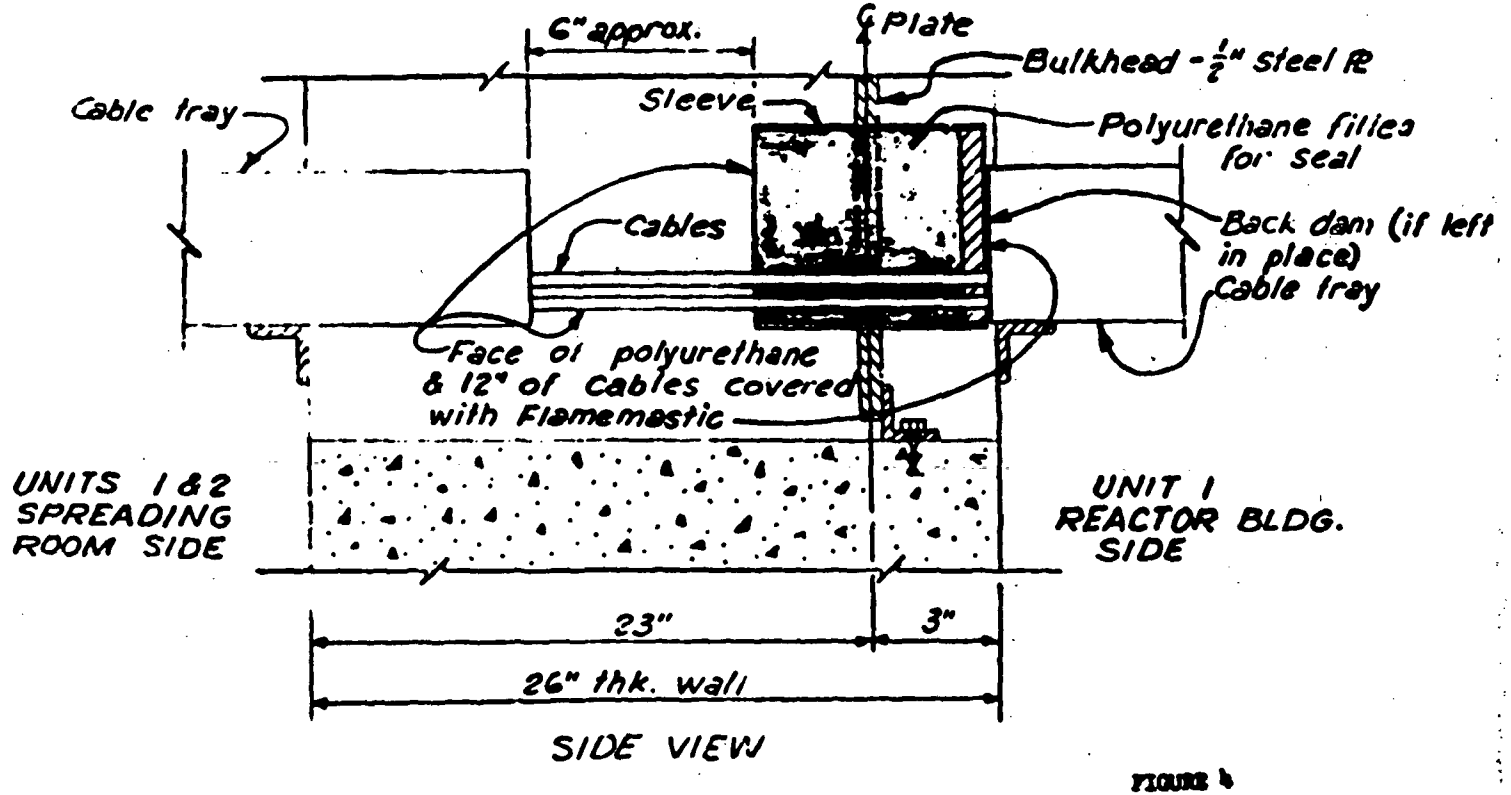


FIGURE 4

PARTIAL CROSS SECTION OF PENETRATIONS
(TO SHOW BULKHEAD LOCATION IN WALL AS CONSTRUCTED)

FIG. 4

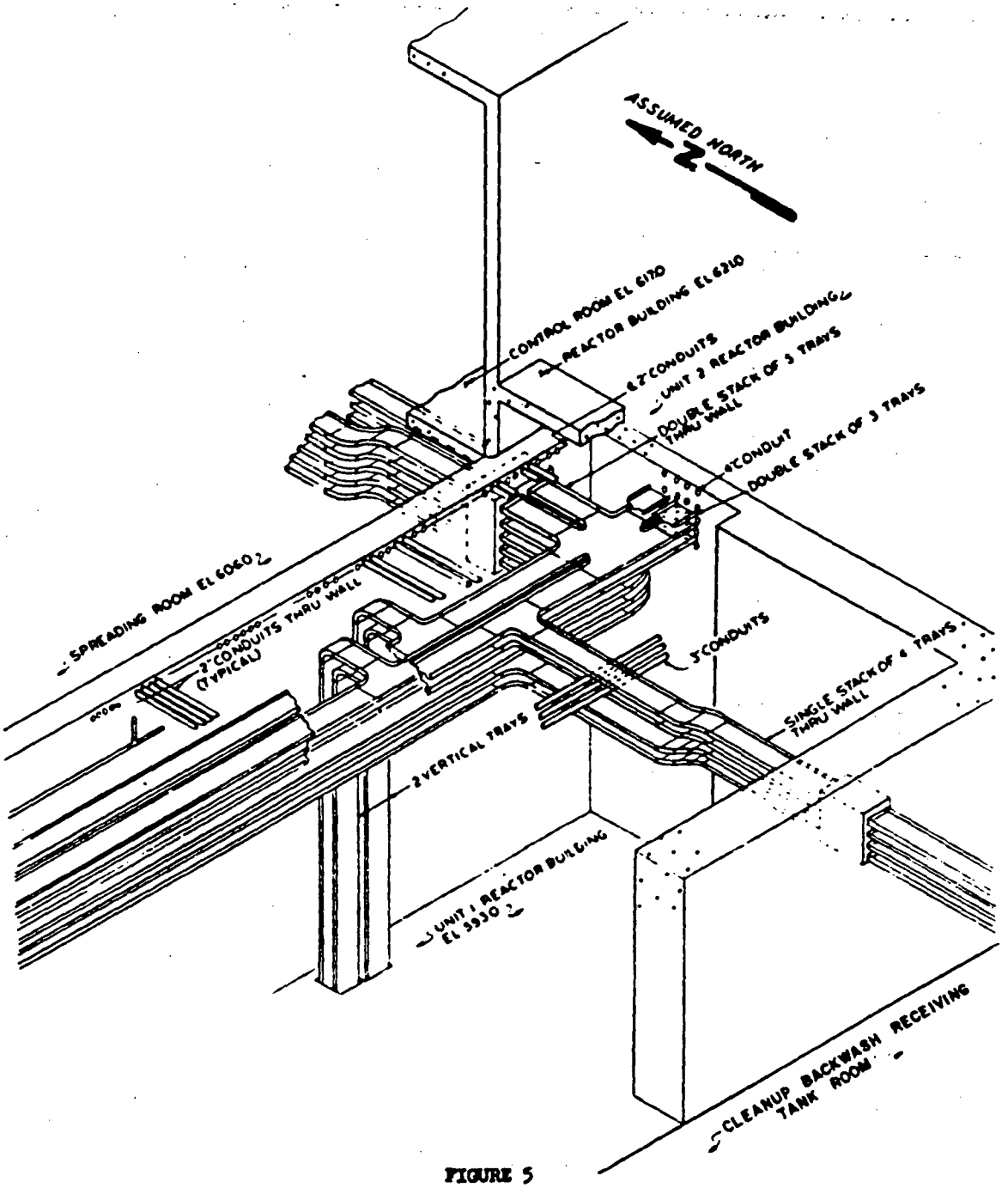


FIGURE 5

AREA OF FIRE

FIG. 5

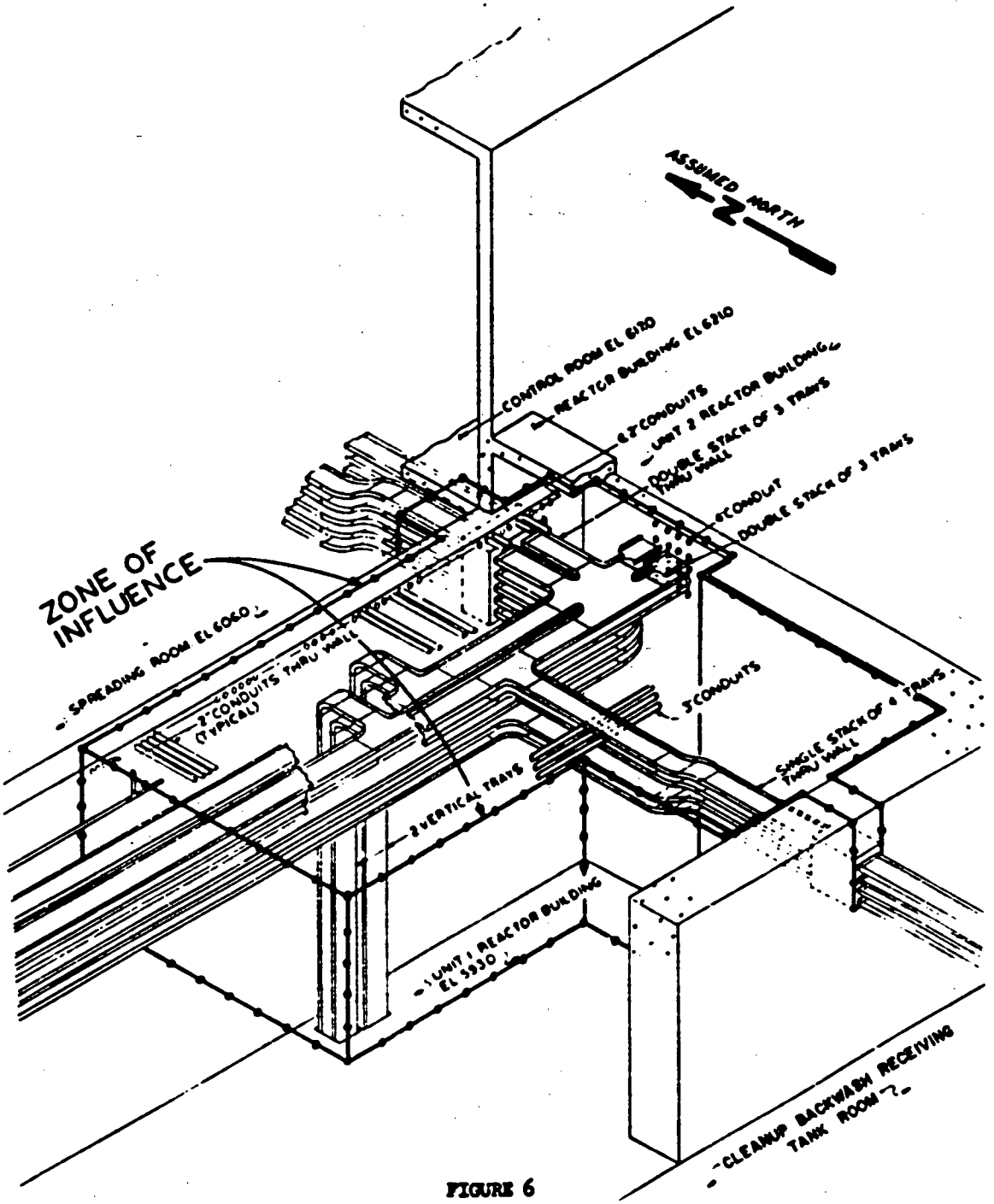


FIGURE 6

AREA OF FIRE

FIG. 6

FIGURE 7

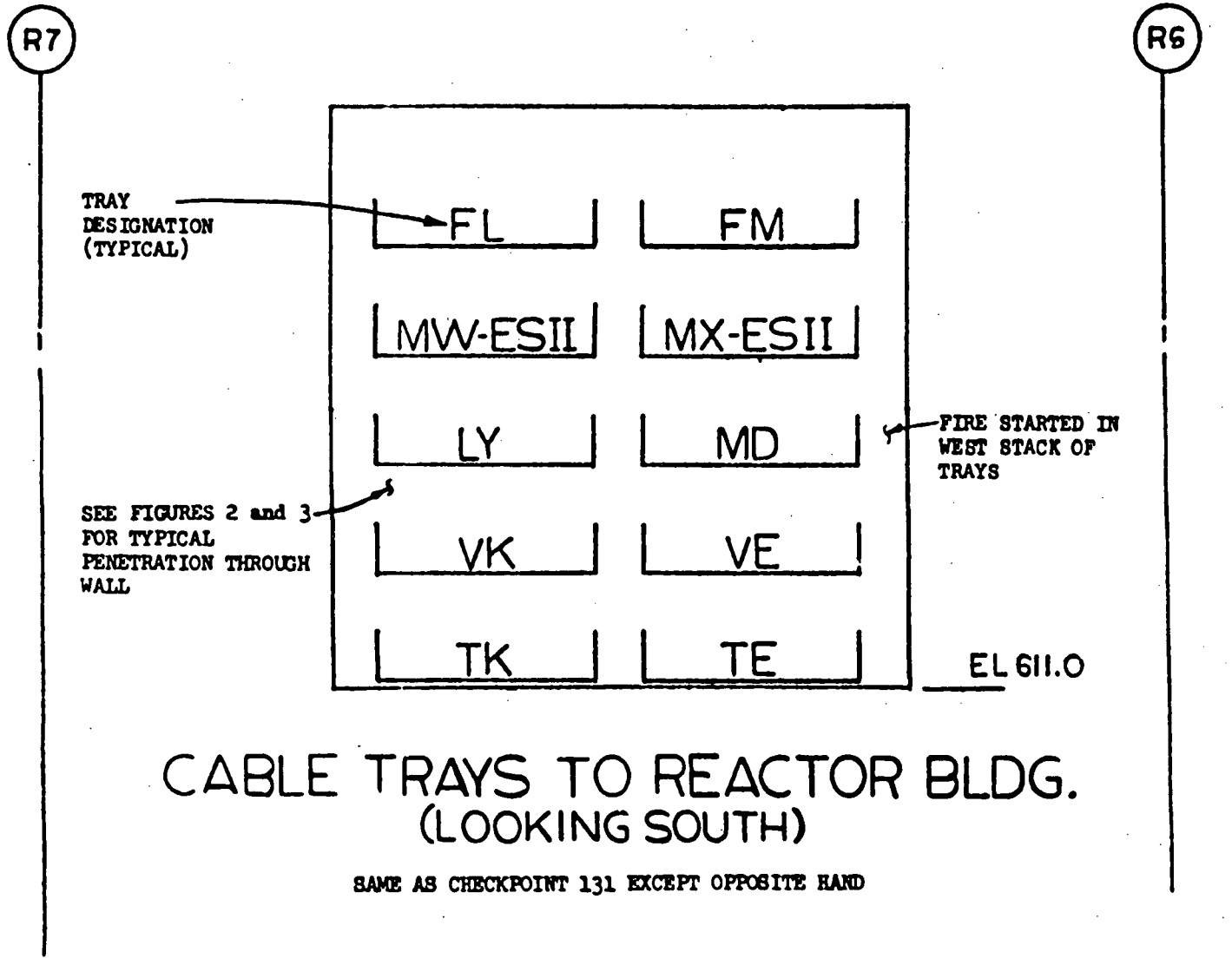


FIG. 7

FIGURE 8
PART PLAN VIEW OF CABLE TRAYS

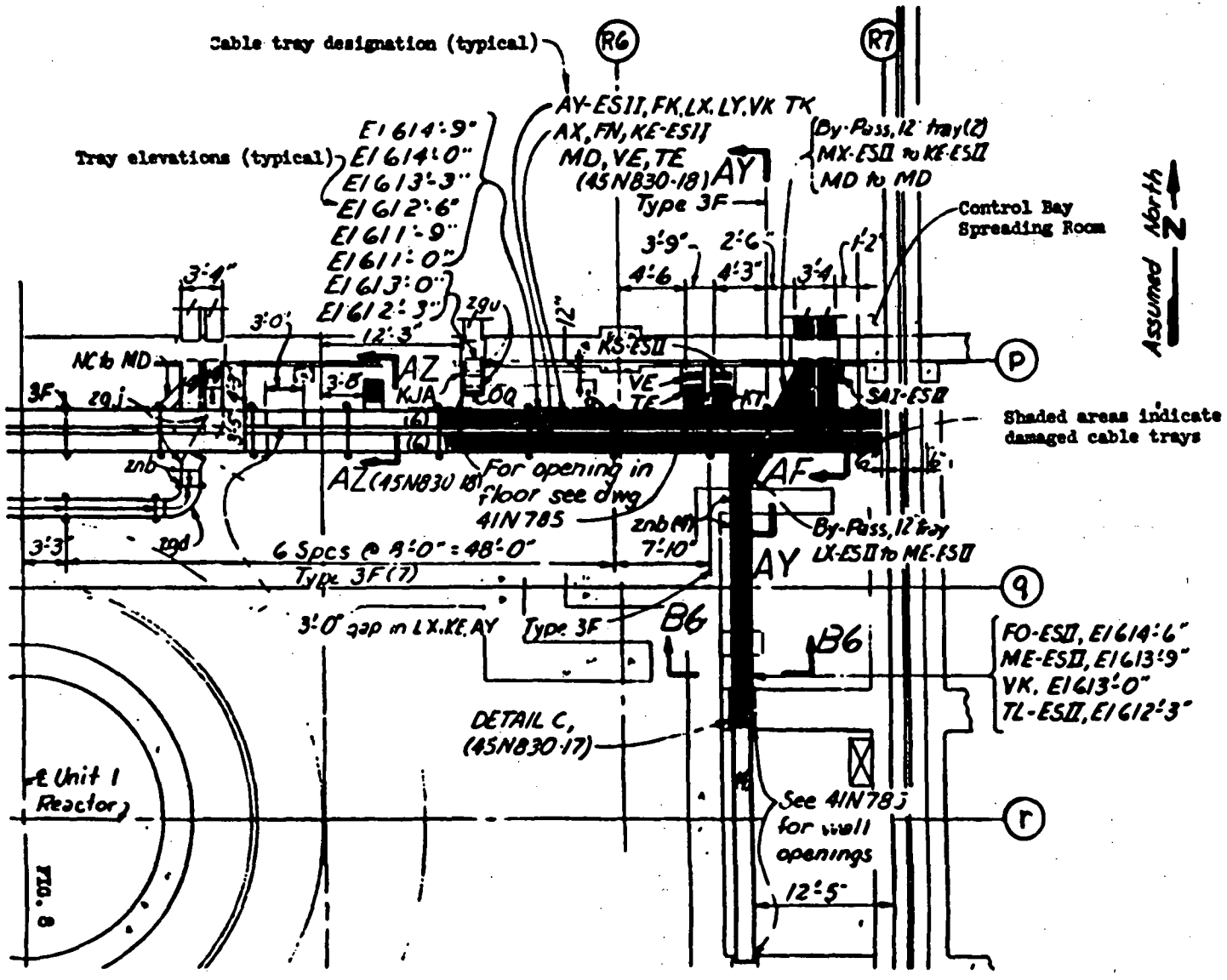


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FIGURE 9
CABLE TRAY SINGLE LINE

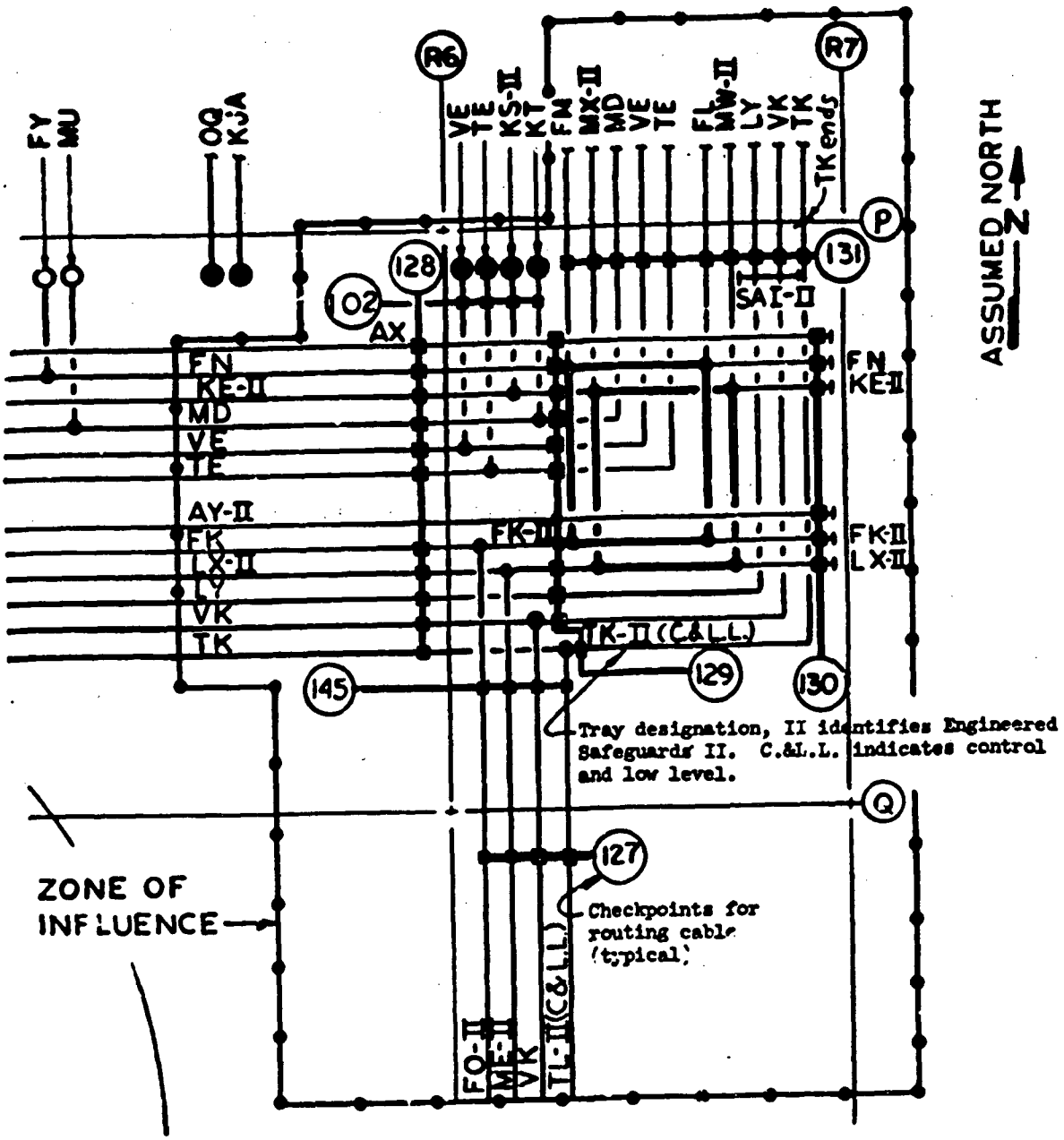

FIG. 9

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FIGURE 10
PART PLAN VIEW OF CONDUITS NEAR TRAYS

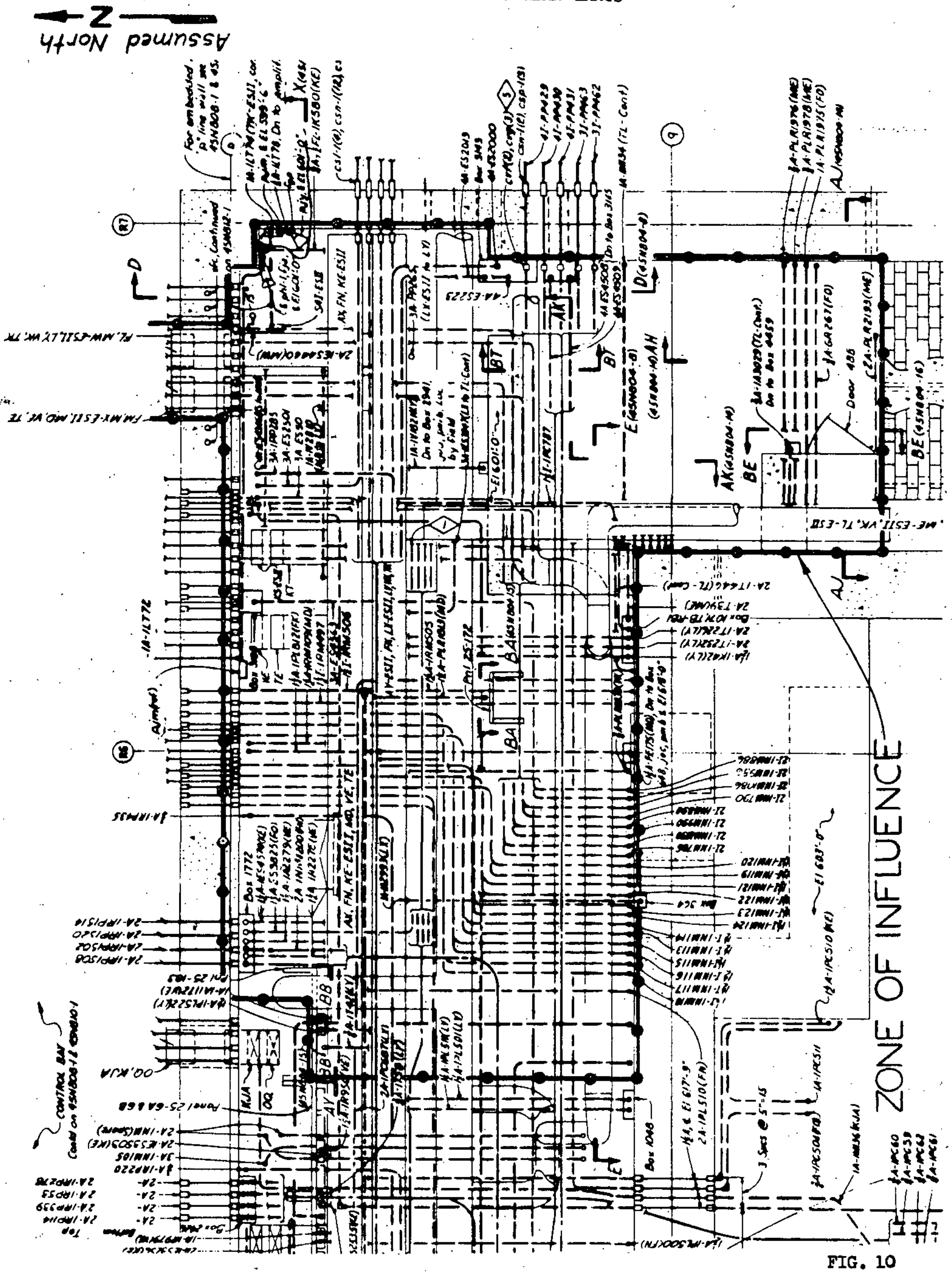


FIG. 10

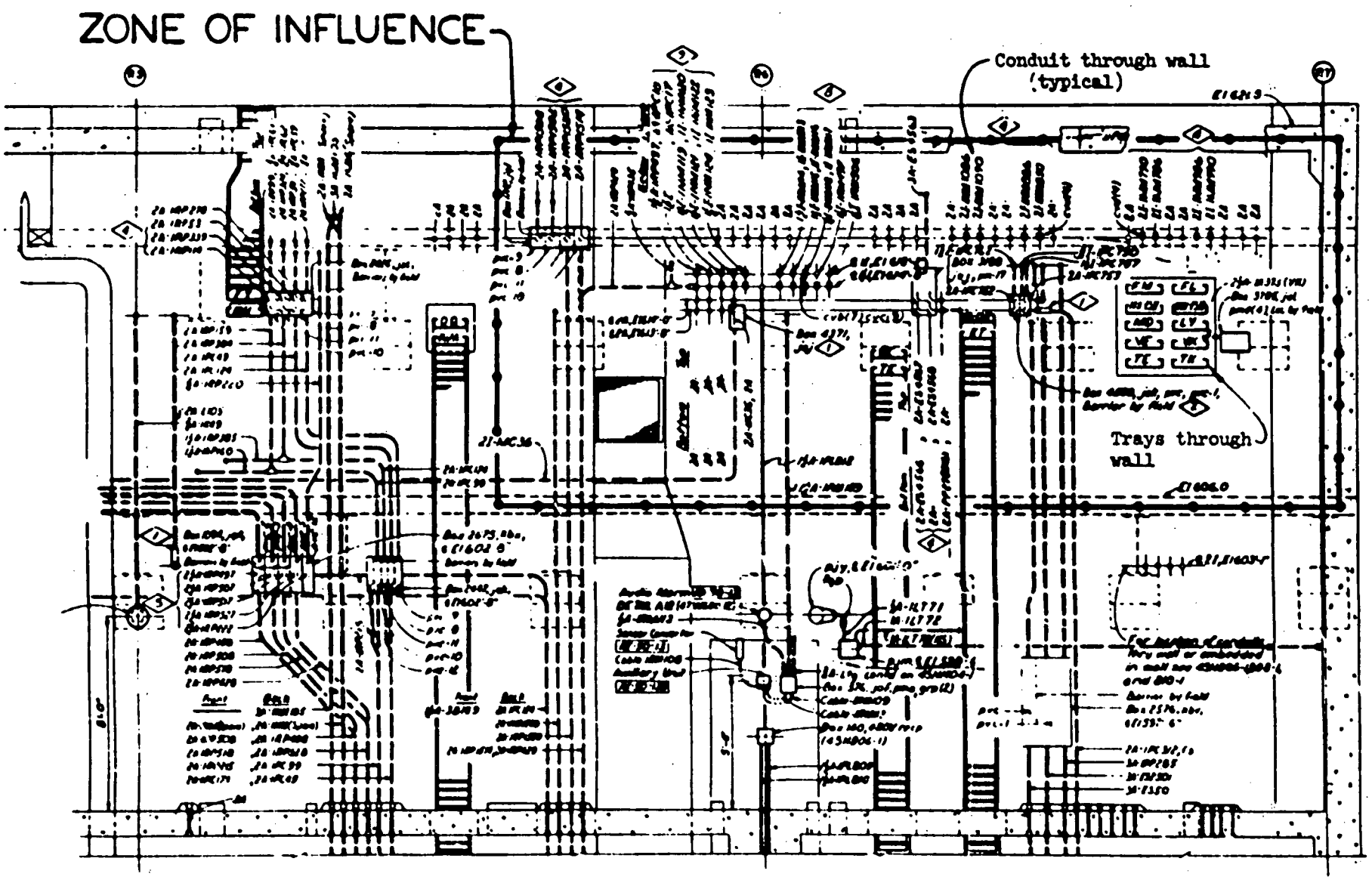


FIGURE 11
ELEVATION VIEW LOOKING NORTH TOWARD CONTROL BAY
FROM REACTOR BLDG UNIT 1 EL 593 SHOWING CONDUITS
AND TRAYS IN ZONE OF INFLUENCE

FIG. 11

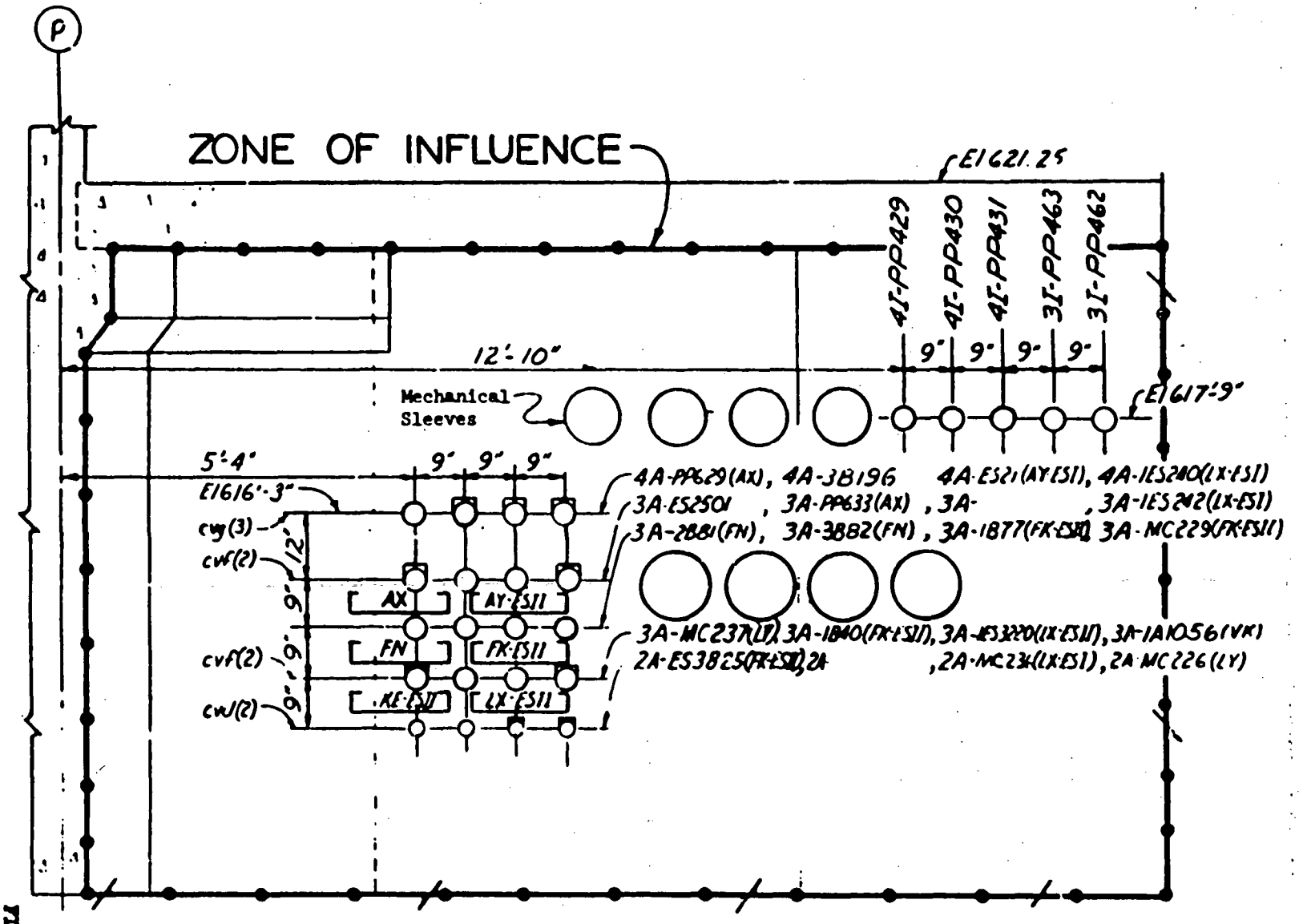


FIG. 12

ELEVATION VIEW LOOKING EAST TOWARD UNIT 2
(cable trays run to wall and stop and cables are fed into unit 2 through conduits.)

FIGURE 12

DESCRIPTION OF SPECIALTY ITEMS ASSOCIATED WITH PENETRATIONS*

	Item	Description	Manufacturer
1.	Froth Pak	Insta-Foam	Insta-Foam Products Company Joliet, Illinois

Froth Pak Insta-Foam is the trade name for a kit using an aerosol dispensing unit which contains the chemical components for making rigid polyurethane foam. When the unit is activated, high-quality froth foam is dispensed from two pressurized containers, forming a rigid cellular polyurethane product in less than 1 minute.

2.	Polyurethane	Pourable type Part A No. 0293A	Witco Chemical Company New Castle, Delaware
		Pourable type Part B No. 67010	

Polyurethane, pourable type, produces a rigid cellular polyurethane product similar to that produced by the Froth Pak Insta-Foam. The liquids, part A and part B, are mixed equally by pouring back and forth between two containers until mixed and reaction starts. Before it expands, it more readily flows into small crevices to effect a better seal upon expansion.

3.	Flamemastic	71A	Dyna-Therm Corporation 598 West Avenue Los Angeles, California

Dyna-Therm Flamemastic coatings are compounded of thermoplastic resinous binders, flame-retardant chemicals, and inorganic incombustible fibers. They have a gray fibrous appearance when dry.

4.	Marinite panels	No. 36, type B	Johns-Manville

Marinite panels are composed of incombustible asbestos fibers, diatomaceous silica, and a hydrothermally-produced inorganic binder. They were originally developed to isolate and prevent the spread of shipboard fires. They are hard, dense boards.

5.	Resilient polyurethane foam		Hickory Springs Manufacturing Company 2200 Main Avenue, SE. Hickory, North Carolina

Resilient polyurethane foam is a preformed, resilient, cellular polyurethane foam material which was developed primarily to make furniture cushions.

6.	Styrofoam	---	Unknown

Styrofoam is a lightweight, preformed thermal-insulating material and packing material. It is commonly used for making ice chests. It is readily found on construction sites since it is also used as protective packing material for fragile equipment.

TABLE 1
SHEET 1 OF 2

Item	Description	Manufacturer
7. RTV 102 white	Silicone rubber	General Electric Company Silicone Products Department Waterford, New York

RTV (room temperature vulcanizing) silicone rubber is a liquid "rubber" (not a natural rubber) which cures at room temperature to a resilient, tough adhesive. It was originally developed for sealing space vehicles. It is commonly used in the home to seal around bathtubs.

8. Ty-Rap cable ties	TY-525M	Thomas and Betts Elizabeth, New Jersey

Ty-Rap cable ties are small straps about 1/32 inch thick and 1/8 inch wide, of varying lengths, with a loop in one end for binding cables together. They are generally made of nylon or similar plastic.

9. Other materials may have been used in construction penetration seals.

*These "descriptions" are provided by the committee to assist the layman in understanding the various materials. The descriptions should not be construed as definitions or precise technical descriptions.

Checkpoint 102
(Looking North)

VE

TE

KS-ESII

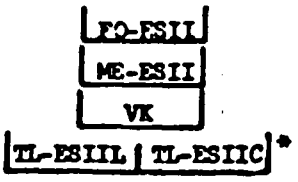
KT

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	67	.353	6.566
	WVA-1	38	.333	3.306
	WVB	3	.371	.324
	WVC	14	.401	1.764
	MFR	1	.242	.046
TOTAL		123		12.006
TE	WUB	117	.231	4.914
	WUB-1	25	.339	2.250
TOTAL		142		7.164

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
KS-ESII	WFB	1	.490	.189
	WFE	2	.659	.684
	WGB	40	.425	5.68
	WGC	3	.444	.465
	WGD	2	.484	.368
	WGE	1	.559	.246
	WGG	3	.660	1.026
	WGI	1	.710	.396
	WGK	1	.789	.490
	WHB	6	.384	.696
	WHC	2	.405	.258
	WHE	4	.480	.608
	WHG	1	.519	.212
	WHI	3	.640	.966
	WHJ	1	.710	.396
	WLB	1	.509	.204
TOTAL		72		12.884
KT	WGB	1	.425	.142
	WGG	1	.660	.342
	WHB	2	.384	.232
	WHD	7	.439	1.064
	WHG	1	.519	.212
	WTO	1	.340	.091
	WTR	1	.360	.102
	WVE	1	.461	.167
	WVG	1	.587	.271
	WVI	11	.834	7.205
	WVJ	8	1.012	6.440
	WVR	101	.650	33.532
	WVU-1	2	.439	.304
TOTAL		138		50.104

TABLE 2
SHEET 1 OF 11

Checkpoint 127
(Looking North)



TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	4	.429	.580
	WDG	7	.485	1.295
	WDH	3	.619	.903
	WDI	3	.660	1.026
	WLB	1	.509	.204
	WLC	4	.539	.916
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		30		6.678
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFB	1	.490	.189
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU-1	2	.439	.304
	WWN	2	.0172	.0344
TOTAL		137		33.530

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	8	.353	.784
	WVA-1	20	.333	1.740
	WVB	9	.371	.972
TOTAL		37		3.496
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	21	.425	.284
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHD	1	.439	.152
	WHE	1	.480	.181
	WHG	4	.519	.848
	WTO	9	.340	.819
	WTR	1	.360	.102
	Belden 8213	1	.405	.129
TOTAL		70		7.396
TL-ESIIL	WUB	40	.231	1.680
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		52		2.824

* C - indicates control level portion of TL
L - indicates low level portion of TL

TABLE 2
SHEET 2 OF 11

Checkpoint 128
(Looking East)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	WHB	6	.915	3.942
FN	WLC	1	.539	.229
	WLS	1	.817	.524
	WDG	12	.485	2.220
	WDN	13	.940	9.022
	WGD	5	.484	.920
TOTAL		32		12.915
KE-ESII	WHE	2	.480	.362
	WHJ	3	.710	1.188
	WDD	2	.340	.182
	WFE	1	.659	.342
	WGB	4	.425	.568
	WGG	1	.660	.342
	WHB	22	.384	2.552
	WWN	1	.148	.0172
TOTAL		36		5.708
MD	WHD	7	.439	1.064
	WHE	1	.480	.181
	WHG	1	.519	.212
	WGB	9	.425	1.278
	WGE	1	.559	.246
	WGG	1	.660	.342
	WHB	2	.384	.232
	WVA-1	2	.333	.174
	WVI	1	.834	.655
	WVJ	8	1.012	5.240
	WVU-1	1	.439	.152
TOTAL		34		9.776

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0			
FK	WDF	8	.429	1.160
	WDG	10	.485	1.850
	WDH	5	.619	1.505
	WDI	3	.660	1.026
	WDK	1	.769	.465
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
TOTAL		41		15.268
LX-ESII	WHC	1	.405	.129
	WHE	1	.420	.181
	WTO	7	.340	.637
	WGC	1	.444	.155
	WGD	1	.484	.184
	WGE	2	.559	.492
	WGG	1	.660	.342
	WGI	1	.710	.396
	WGK	2	.789	.980
TOTAL		17		3.496
LY	WTD	1	.638	.320
	WTO	24	.340	2.184
	WTR	2	.360	.204
	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	4	.600	1.132
	WGB	10	.425	1.420
	WGC	2	.444	.310
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGG	5	.660	1.710
TOTAL		56		8.819

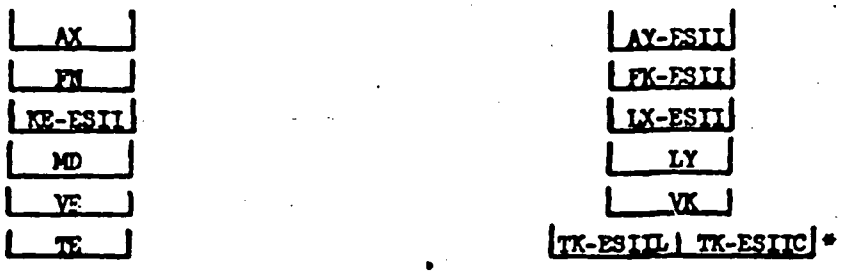
TABLE 2
SHEET 3 OF 11

Checkpoint 128
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VE	WVA	94	.353	9.212
	WVA-1	85	.333	7.395
	WVB	12	.317	1.296
	WVC	4	.401	.504
TOTAL		195		18.407
TE	WUB	80	.231	3.360
	WUB-1	23	.339	2.070
	WVA	3	.353	.294
TOTAL		106		5.724

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
LY	WGK	1	.789	.490
	WHB	6	.384	.696
TOTAL		7		1.186
VK	WVA	29	.353	2.842
	WVA-1	91	.333	7.917
	WVB	9	.371	.972
TOTAL		129		11.731
TK	WUB	16	.231	.672
	WVA	6	.353	.588
TOTAL		22		1.260

Checkpoint 129
(Looking East)



TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	Same as checkpoint 128			
FN	Same as checkpoint 128			
KE-ESII	WDD	2	.340	.182
	WFB	1	.490	.189
	WFE	1	.659	.342
	WGB	43	.425	6.106
	WGC	3	.444	.465
	WGD	6	.484	1.104
	WGE	1	.559	.246
	WGG	4	.660	1.368
	WGI	1	.710	.396
	WHB	26	.384	3.016
	WHC	2	.405	.258
	WHE	4	.480	.724
	WHG	2	.519	.424
	WHI	2	.640	.644
	WHJ	4	.710	1.584
	WLB	1	.509	.204
	WWN	1	.0172	.0172
TOTAL		104		18.024
MD	Same as checkpoint 131			
VE	Same as checkpoint 131			
TE	Same as checkpoint 131			

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII	0	0	0	0
FK-ESII	WDF	10	.429	1.45
	WDG	7	.485	1.295
	WDH	2	.619	.602
	WDK	4	.769	1.860
	WDN	8	.940	5.552
	WDO	4	1.029	3.332
	WFB	2	.490	.378
	WHB	1	.384	.116
	WLB	1	.509	.204
	WLN	1	.559	.245
	WLO	2	.627	.618
	WVA	2	.353	.196
TOTAL		44		14.544
LX-ESII	WDD	4		.364
	WDE	2		.226
	WFB	1	.490	.189
	WGB	25	.425	3.550
	WGD	3	.484	.552
	WGE	5	.559	1.230
	WGG	2	.660	.684
	WGI	36	.710	14.256
	WGK	5	.789	2.450
	WGM	1	.874	.600
	WHB	6	.384	.696
	WHC	9	.405	1.161
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	3	.519	.636
	WHI	4	.640	1.288
	WHJ	7	.710	2.872
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WVU-1	2	.439	.304
	WWN	2	.0172	.034
TOTAL		123		33.718

* C - indicates control level portion of TL
L - indicates low level portion of TL

TABLE 2
SHEET 5 OF 11

Checkpoint 129
(Looking East)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
					TK-ESIIC	WFB	8	.490	1.512
						WFD	1	.600	.283
						WGB	16	.425	2.272
						WGC	6	.444	.930
						WGD	2	.484	.368
						WGE	2	.559	.492
						WGI	2	.710	.792
						WGK	4	.789	1.960
						WHB	4	.384	.464
						WHC	1	.405	.129
						WHD	1	.439	.152
						WHE	2	.480	.362
						WHG	4	.519	.848
						WTO	1	.340	.091
						WTR	1	.360	.102
						Belden 8213	1	.405	.129
					TOTAL		56		10.886
					TK-ESIIL	WHB	1	.384	.116
						WUB	34	.231	1.428
						WUB-1	4	.339	.360
						WVA	4	.353	.392
					TOTAL		43		2.296
					LY	Same as checkpoint 128			
					VK	Same as checkpoint 128			

TABLE 2
SHEET 6 OF 11

Checkpoint 130
(Looking West)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AX	Same as checkpoint 129			
FN	WDN	4	.940	2.576
	WLS	1	.817	.524
TOTAL		5		3.100
KE-ESII	WLB	1	.509	.204
TOTAL		1		.204

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
AY-ESII		0		
FK-ESII	WDF	8	.429	1.160
	WDG	8	.485	1.480
	WDH	2	.619	.602
	WDK	1	.769	.465
	WDN	8	5.552	.940
	WFB	2	.490	.378
	WLO	1	.627	.309
TOTAL		30		5.334
LX-ESII	WGB	4	.425	.568
	WGD	1	.484	.184
	WGG	1	.660	.342
	WGK	2	.789	.980
	WHB	2	.384	.232
TOTAL		10		2.306

TABLE 2
SHEET 7 OF 11

Checkpoint 131
(Looking North)

FM
MX-ESII
MD
VE
TE

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FM	WDF	2	.429	.290
	WDG	12	.485	2.220
	WDK	3	.769	1.395
	WDN	3	.940	2.082
	WDO	4	1.029	3.332
	WLC	1	.539	.229
TOTAL		25		9.573
MX-ESII	WDD	2	.340	.182
	WDE	2	.379	.226
	WFB	1	.429	.145
	WFE	2	.659	.684
	WGB	58	.425	8.236
	WGC	1	.444	.155
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	4	.660	1.026
	WGI	37	.710	14.652
	WGK	3	.789	1.47
	WGM	1	.874	.600
	WHB	26	.384	3.016
	WHC	10	.405	1.290
	WHD	4	.439	.608
	WHE	7	.480	1.267
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	4	.710	1.584
	WHL	1	.781	.479
	WVJ-1	2	.439	.304
	WWN	1	.148	.0172
TOTAL		181		38.290
MD	WGB	1	.425	.142
	WGE	1	.559	.246
	WHB	1	.384	.116
	WHD	2	.439	.304
	WHE	1	.480	.181
	WTO	1	.340	.091
	WTR	1	.360	.102
	WVE	1	.461	.167

FL
MW-ESII
LY
VK
TK

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FL	WDN	6	.940	4.164
	WLB	1	.509	.204
	WLN	1	.559	.245
	WLO	1	.627	.309
TOTAL		9		4.922
MW-ESII	WFB	9	.490	1.701
	WFD	1	.600	.283
	WGB	23	.425	3.266
	WGC	6	.444	.930
	WGD	3	.484	.552
	WGE	2	.559	.492
	WGI	3	.710	1.188
	WGK	3	.789	1.47
	WHB	14	.384	1.624
	WHE	5	.480	.905
	WHJ	1	.710	.396
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	Belden 8213	1	.405	.129
TOTAL		73		11.919
LY	WDD	2	.340	.182
	WFB	1	.490	.189
	WFD	3	.600	.849
	WGB	8	.425	1.136
	WGC	2	.444	.310
	WGD	2	.484	.368
	WGE	5	.559	1.230
	WGG	5	.660	1.710
	WGK	1	.789	.490
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHI	3	.640	.966
	WVA-1	2	.333	.174
TOTAL		41		5.783

TABLE 2
SHEET 8 OF 11

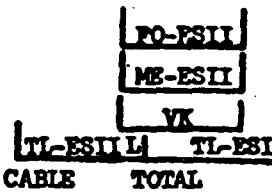
Checkpoint 131
(Looking North)
(Continued)

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
MD (Continued)	WVG	1	.587	.271
	WVI	11	.834	7.205
	WVJ	16	1.012	12.880
	WVR	101	.650	33.532
	WVU-1	3	.439	.456
TOTAL		141		55.693
VE	WVA	52	.353	5.096
	WVA-1	61	.333	5.307
	WVB	15	.371	1.620
	WVC	10	.401	1.260
	MFRS	1	.242	.046
TOTAL		139		13.329
TE	WUB	37	.231	1.974
	WUB-1	47	.339	4.230
	WVA	3	.353	.2940
TOTAL		87		6.498

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	41	.353	3.332
	WVA-1	46	.333	4.002
TOTAL		87		7.334
TK	WUB	34	.231	1.428
	WUB-1	25	.339	2.250
	WVA	2	.353	.196
TOTAL		61		3.874

Checkpoint 145

(Looking North)



TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
FO-ESII	WDD	3	.340	.273
	WDF	2	.429	.290
	WDG	6	.485	1.110
	WDH	1	.619	.301
	WDI	1	.660	.342
	WLB	1	.509	.204
	WLC	1	.539	.229
	WLN	1	.559	.245
	WLO	4	.627	1.236
TOTAL		20		4.230
ME-ESII	WDE	2	.379	.226
	WDD	4	.340	.364
	WFD	1	.600	.283
	WGB	27	.425	3.834
	WGD	3	.484	.552
	WGE	4	.559	.984
	WGG	1	.660	.342
	WGI	35	.710	13.860
	WGK	3	.789	.147
	WGM	1	.874	.600
	WHB	5	.384	.580
	WHC	8	.405	1.032
	WHD	4	.439	.608
	WHE	5	.480	.905
	WHG	4	.519	.848
	WHI	4	.640	1.288
	WHJ	7	.710	2.772
	WHL	1	.781	.479
	WLB	1	.509	.204
	WTA	1	1.139	1.020
	WTO	8	.340	.728
	WTR	1	.360	.102
	WVU	2	.376	.222
	WWN	2	.0172	.034
TOTAL		134		32.014

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
VK	WVA	10	.353	.980
	WVA-1	22	.333	1.914
	WVB	9	.371	.972
TOTAL		41		3.866
TL-ESIIC	WFB	8	.490	1.512
	WFD	1	.600	.283
	WGB	29	.425	4.118
	WGC	5	.444	.775
	WGD	1	.484	.184
	WGE	4	.559	.984
	WGI	2	.710	.792
	WGK	3	.789	.147
	WHB	6	.384	.696
	WHC	1	.405	.129
	WHD	1	.439	.152
	WHE	2	.480	.362
	WHG	4	.519	.848
	WTO	8	.340	.728
	WTR	1	.360	.102
	BELDEN 8213	1	.405	.129
TOTAL		77		11.941
TL-ESIIL	WUB	50	.231	2.100
	WUB-1	4	.339	.360
	WVA	8	.353	.784
TOTAL		62		3.244

* C - indicates control level portion of TL
L - indicates low level portion of TL

TABLE 2
SHEET 10 OF 11

Note: Tray loading for vertical tray connecting trays MW-II and TK-II south of checkpoint 131

SAI-ESII

TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA	TRAY DESIG	CABLE TYPE	QTY	CABLE OD	TOTAL AREA
SAI-ESII	WFB	8	.490	1.512					
	WFD	1	.600	.283					
	WGB	19	.425	2.698					
	WGC	4	.444	.620					
	WGD	2	.484	.368					
	WGE	2	.559	.492					
	WGI	2	.710	.792					
	WGK	3	.789	1.470					
	WHB	6	.384	.696					
	WHC	1	.405	.129					
	WHD	1	.439	.152					
	WHE	2	.480	.362					
	WHG	4	.519	.848					
	Belden 8213	1	.405	.129					
TOTAL		56		10.551					

COMPUTED BY CABLE TRAY REV FO-ESII CHECKED BY BU-BU DATE 128 & 145

CABLE	PURPOSE	FROM	TO	SCHEM	TYPE	WHERE TO DISCONNECT OR DE-ENERGIZE AND OTHER COMMENTS (40)
B331	Diesel Gen Air Compr Back up Mot Alternate feed	Batt Bd 1 Pnl 7 Bkr 712 45N701-1	DC Transfer Sw 45N786-11	45N774-6	wdg	Open breaker 712, BATT BD 1, panel 7
ES3825-II	Battery & Bd Rm exhaust fan 1B Supply	480 Cont Bay Vent Bd 1B Pnl 6C 45N788-2	Fan Motor 1B 45N788-4	45N769-7	wlo	Open breaker compartment 6C, 480V cont. Bay Vent Bd B
ES3900-II	Bd Rm Emer Supply fan 1B Supply	480V Reac MOV Bd 1B Pnl 17A 45N1750-6	Fan Motor 45N788-4	45N769-7	wld	Open breaker compartment 17A, 480V REAC MOV Bd 1B
PL478	Reactor Bldg elevator No. 1 supply	Common Bd Pnl 7B 45N781-2	Mtr's JB 45N781-2	45N769-1	wdg	Open breaker compartment 7B, 480V common Bd 1
PLRA75	Clean-up Backwash Transfer Pump 1A Supply	480V R.B. Vent Bd 1B Pnl CA 45N1756-2	Pump Motor 45N1756-9	45N777-5	wlo	TRIP BKR 480V REACT BLDG VENT BD 1B, COMPT 6A
K321	SEE SHEET 9					
1B89-1E	Alternate Supply, 250V CONTROLS	Batt Bd 2 Pnl 7 Bkr 708 45N702-1	480V Shutdn Bd 1A Pnl 4 45N1748-2	45N769-1	wdt	TRIP BKR 708, BATT BD 2, PNL 7
1B93-1E	Alternate Supply, 250V CONTROLS	Batt Bd 3 Pnl 7 Bkr 709 45N703-1	480V Shutdn Bd 1B Pnl 4 45N1748-3	45N749-2	wdt	TRIP BKR 709, BATT BD 3, PNL 7
1ES3000-II	Core Spray Sys II outboard ditch valve (FCV-75-53) Supply	480V R. MOV Bd 1B Pnl 7E 45N1750-3	FCV-75-53 45N1750-11	45N751-3	wlo	TRIP BKR 480V REACT MOV BD 1B, COMPT 7E
1ES3013-II	Core Spray Sys II outboard disch valve (FCV-75-51) Supply	480V Reac MOV Bd 1B Pnl 8E 45N1750-3	FCV-75-51 45N1750-11	45N751-3	wlo	TRIP BKR 480V REACT MOV BD 1B, COMPT 8E
1K575	Evacuation Alarm System	480V fuse Pnl A 45N785-12	Local Starter in JB 2826 55N2779	55N2779	wlb	PULL FU 2AX & 2AY EVAC ALARM FUSE PNL A EL 606, COL PR11
1K606	Evacuation Alarm system	480V fuse Pnl B 45N786-12	Local Starter in JB 2826 55N2779	do	wln	PULL FU 1BX & 1BY EVAC ALARM FUSE PNL B EL 606, COL PR12
1PCS04-II	Reactor water Clean-up Sys inst valve (FCV-69-12) supply	480V Reactor MOV Bd 1B Bd 17E 45N1750-6	FCV-69-12 45N1750-7	45N751-3	wlc	TRIP BKR 480V REACT MOV BD 1B, COMPT 17E
1PL474	Unit Preferred MMG Set 1. nor fdr supply	480 Shutdn Bd 1A Pnl 7D 45N1748-1	Unit Preferred MMG Set 1 Control Cab. 45N708-2	45N749-1	wdh	TRIP BKR 480V SHTDN BD 1A COMPT 7D

SAMPLE CABLE TABULATION SHEET

TABLE 3
SHEET 1 OF 1

Number of each class of safety related cables routed in fire zone.

Plant Usage	Number	Safety Classification	Channel or Division*
Common	20	Engineered Safeguard - ECCS	I
	20	Engineered Safeguard - ECCS	II
	13	Engineered Safeguard - Diesel A	IA
	33	Engineered Safeguard - Diesel C	IIC
	5	Engineered Safeguard - Diesel D	IID
	7	Load Shedding - Diesel A	A1
	9	Load Shedding - Diesel C	B1
	7	Supporting Auxiliaries - Electrical	IE
Subtotal	114		
Unit 1	6	Engineered safeguard - ECCS	I
	182	Engineered Safeguard - ECCS	II
	4	Load Shedding - Diesel A	A1
	5	Load Shedding - Diesel C	B1
	1	Load Shedding - Diesel D	B2
	52	Neutron Monitoring (also activates	IA
	52	Neutron Monitoring RPS)	IB
	52	Neutron Monitoring	IIA
	52	Neutron Monitoring	IIB
	14	Primary Containment Isolation	I
	39	Primary Containment Isolation	II
	2	Reactor Protection (control rod	IA
	2	Reactor Protection scram)	IB
	2	Reactor Protection	IIA
	2	Reactor Protection	IIB
	3	Reactor Protection	IIIB
	12	Supporting Auxiliaries - Electrical	IE
Subtotal	482		
Unit 2	15	Engineered Safeguard - ECCS	I
	3	Engineered Safeguard - ECCS	II
	4	Supporting Auxiliaries - Electrical	IE
Subtotal	22		
Unit 3	4	Engineered Safeguards - ECCS	I
	3	Engineered Safeguards - ECCS	II
	3	Supporting Auxiliaries - Electrical	IE
Subtotal	10		
TOTAL	628		

*See sheets 2 & 3 for channel or division definitions.

TABLE 4
SHEET 1 OF 3

SUMMARY OF CABLE SUFFIX DEFINITIONS

1. Cables in the PP series with an A or B suffix are to be separated from each other. They are not engineering safeguard cables, but a separate routing is desirable. They involve off-site power.

2. The following suffixes apply to all cable series:

   I - Division I engineering safeguard or Primary Containment Isolation cables

   II - Division II engineering safeguard or Primary Containment Isolation cables

   IA - Diesel generator A shutdown logic cables (may be routed in cable tray with Division I cables)

   IB - Diesel generator B shutdown logic (routed in conduit)

   IIC - Diesel generator C shutdown logic (may be routed in cable tray with Division II cables)

   IID - Diesel generator D shutdown logic cables (routed in conduit)

3. The following suffixes apply to LS series:

   A1 - 480V load shedding logic channel A1: (routed with IA-Diesel A)

   A2 - 480V load shedding logic channel A2: (routed with IB-Diesel B)

   B1 - 480V load shedding logic channel B1: (routed with IIC-Diesel C)

   B2 - 480V load shedding logic channel B2: (routed with IID-Diesel D)

4. The following suffixes apply to RP (Reactor Protection) or NM (Neutron Monitoring) series:

   IA - RPS logic channel A1

   IIA - RPS logic channel A2

   IB - RPS logic channel B1

   IIB - RPS logic channel B2

5. The following suffixes apply to RP (Reactor Protection) series:

   IIIA - RPS manual and back-up scram solenoid channel A.

   IIIB - RPS manual and back-up scram solenoid channel B

   A - 120V a-c RPS channels A1, A2, and A3 supply (RPS MG set A)

   B - 120V a-c RPS channels B1, B2, and B3 supply (RPS MG set B)

   G1 - RPS scram solenoid Group 1

G2 - RPS scram solenoid Group 2

G3 - RPS scram solenoid Group 3

G4 - RPS scram solenoid Group 4

6. Suffix IE - Applies to supporting auxiliaries needed for safe shutdown of plant.

Summary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	DESCRIPTION INSULATED*	NO. CABLES DAMAGED
WBB	1/c # 12	1	11+
WCA	1/c # 14	1	2+
WDD	1/c # 8	2	4+
WDE	1/c # 6	2	1+
WDF	1/c # 4	2	7+
WDG	1/c # 2	2	13+
WDH	1/c #1/0	2	2+
WDI	1/c #2/0	2	2+
WDK	1/c #4/0	2	2+
WDN	1/c #300	2	6+
WDO	1/c #400	2	1+
WFB	2/c # 10	3	16
WFC	3/c # 10	3	1
WFD	4/c # 10	3	6
WFE	5/c # 10	3	1
WGB	2/c # 12	3	157
WGC	3/c # 12	3	18
WGD	4/c # 12	3	37
WGE	5/c # 12	3	13
WGG	7/c # 12	3	14
WGI	9/c # 12	3	45
WGK	12/c # 12	3	18
WGM	16/c # 12	3	2
WGN	19/c # 12	3	2
WHB	2/c # 14	3	62
WHC	3/c # 14	3	15
WHD	4/c # 14	3	18
WHE	5/c # 14	3	13
WHG	7/c # 14	3	10
WHI	9/c # 14	3	8
WHJ	9/c # 14	3	7
WHL	16/c # 14	3	1
WHT	7/c # 16	3	8

* Numbers listed correspond to insulation of cable type as shown below.

\+ Number of individual cable designations. Actual number of conductors appear on checkpoint sheets showing tray fill.

mary of cable types involved in fire.

CABLE TYPE (MARK)	DESCRIPTION NO. & SIZE OF CONDUCTORS	DESCRIPTION INSULATED*	NO. CABLES DAMAGED
WLB	2/c # 12	4	8
WLC	3/c # 12	4	5
WLG	7/c # 12	4	1
WLN	2/c # 10	4	1
WLO	3/c # 10	4	5
WLS	7/c # 10	4	1
WMB	1/c #2/0	5	6+
WMC	1/c #4/0	5	2+
WMF	1/c #500	5	3+
WTA	50 pr #19	6	1
WTD	12 pr #19	6	1
WTJ	COAX	7	22
WTK	COAX	8	212
WTK-1	COAX	9	8
WTK-2	COAX	10	4
WTO	2 pr #18	11	42
WTR	2 pr #14	12	4
WUB	Thermocouple	13	167
WUB-1	Thermocouple	14	51
WVA	2/c # 16	15	154
[illegible]-1	2/c # 18	15	206
[illegible]	3/c # 16	15	33
WVC	4/c # 16	15	10
WVE	7/c # 16	15	1
WVG	12/c # 16	15	1
WVI	27/c # 16	15	11
WVJ	37/c # 16	15	16
WVR	29/c # 20	16	101
WVU	5/c # 18	16	4
WVU-1	8/c # 18	16	5
BELDEN 8213	COAX	7	1
MFR'S TV CABLE (BELDEN 8212)	COAX	7	1

* Numbers listed correspond to insulation of cable type as shown below.

\+ Number of individual cable designations. Actual number of conductors appear on checkpoint sheets showing tray fill.

1. Single conductor power or control cable with polyethelene insulation and a nylon jacket over the polyethelene. (Termed "PN" per TVA Specification)

TABLE 5
SHEET 2 OF 4

2. Single conductor power or control cable with cross-linked polyethelene insulation and a polyvinyl-chloride insulation jacket over the cross-linked polyethelene. (Termed "CPJ" per TVA Specification)

3. Multiple-conductor cable with a core of the specified number of single conductors as in 1 above covered by a polyvinyl-chloride outer jacket. (Termed "PNJ" per TVA Specification)

4. Multiple-conductor cable with a core of the specified number of single conductors as in 2 above covered by a polyvinyl-chloride outer jacket. (Termed "CPJJ" per TVA Specification)

5. Single conductor high-voltage (5000 volts) power cable with extruded stand and cross-linked polyethelene insulation with metallic electrostatic shielding and polyvinyl chloride jacket overall. (Termed "CPSJ" per TVA Specification)

6. Telephone cable with high density polyethelene over each conductor, mylar backed rubber cable tape, aluminum shield, and high density polyethelene jacket overall. Some of these had polyvinyl chloride jacket overall.

7. Coaxial signal cable with both conductor and overall jacket insulated with polyethelene.

8. Coaxial signal cable with conductor insulated with polyethelene and polyvinyl chloride jacket overall.

9. Coaxial signal cable with conductor and overall jacket insulated with irradiated blend of polyolefins and polyethelene and noise free. Some of these types had cross-linked polyethelene over both.

10. Same as 8 but made noise free by a carbon suspension.

11. Same as 6 except without shield.

12. Twisted pair cable with polyethelene over each conductor and polyvinyl chloride jacket overall.

13. Thermocouple cable with high density polyethelene over each conductor, aluminum foil/mylar type laminated shield, and high density polyethelene overall.

14. Thermocouple cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape shield, and chlorosulfonated polyethelene jacket overall.

15. Signal cable with heat and light stabilized cross-linked polyethelene over each conductor, aluminum foil/mylar tape laminated shield, fiberglass reinforced silicone tape assembly wrap, and chlorosulfonated polyethelene jacket overall.

16. Multiple-conductor cable with core of specified number of single conductor cables insulated with cross-linked polyethelene and a neoprene jacket overall.

BROWNS FERRY UNIT 1

SEQUENCE OF
SIGNIFICANT OPERATIONAL EVENTS
AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1235	Initial Condition	Routine Operation	Unit Load 1,100 MWe
1235	Report of fire received by assistant shift engineer from public safety officer.	Assistant shift engineer set off fire alarm and proceeded to fire. Fire alarm sealed in by unit operator who then used paging system to inform plant personnel of fire location.	Operating personnel fire brigade reported to fire and began fire-fighting activities (describe elsewhere in investigation report).
~ 1240	Received the following alarms in unit 1 control room: 1. RHR or core spray pumps running/ auto blowdown permissive 2. Reactor level low/ to blowdown permissive 3. Core cooling system/diesel initiate	Unit operator observed control board and determined normal reactor water level and steam pressure, drywell pressure normal at 0.45 psig, and emergency core cooling system (ECCS) equipment aligned in normal standby status. (Reactor water level instrumentation activates the emergency core cooling systems, this being normal, indicated a lack of need for these systems.) (Normal drywell pressure indicates that piping was intact inside the primary containment.)	All diesel generators (D/G's) started from ECCS logic signal which started the core spray pumps.
~ 1242	Residual heat removal (RHR) and core spray (CS) pump running alarm received. High-pressure coolant injection pump (HPCI), reactor core isolation coolant pump (RCIC) started.	Unit operator observed pumps running and RHR aligned to reactor in low-pressure coolant injection (LPCI) mode. Verified reactor water level normal and stopped pumps. Operator attempted to reset alarm. (All four of these systems are ECCS and with normal level were not required.)	Pumps stopped. Alarm would not reset with reactor pressure and level normal.
~ 1244	RHR and core spray pumps restarted with no apparent reason.	Operator observed reactor level normal and attempted to stop RHR and core spray pumps. Pumps could not be stopped from benchboard.	Operator did stop pumps at ~ 1248 from benchboard.
~ 1248	Reactor recirculation pumps run back for no apparent reason. Began losing electrical boards. Indicating lights over valve and pump control switches on panel 9-3 were glowing brightly, dimming, and going out. (Panel 9-3 is the control board location for all ECCS equipment.) The lights being lost on control circuits for ECCS pumps and valves precluded reliable operation from that control board.	Operator observed reactor power decreasing and average power range monitors (APRM) responding. Also noted reactor level 2 to 3 inches high. Operators observed smoke from control wiring under panel 9-3.	Unit power decreased from 1,100 MWe to 700 MWe.

778.

Time	Event	Action	Response or Nonresponse
~ 1248 (Contd.)	Lost 1/2 of reactor protection system (RPS). Lost remote manual control of a number of relief valves. Numerous alarms occurred on all control panels and unit in unstable swing.		
~ 1251	Shift engineer instructed operator to zero recirculating pump loading and scram the reactor.	Operator reduced loading signal to recirculating pumps and manually scrammed the reactor and placed reactor mode switch in the shutdown position. Shift engineer reported plant conditions to supervisors by phone.	Recirculating pumps tripped by unknown cause at approximately 20 percent loading. Reactor scrammed and all rods inserted.
~ 1253	Confirmed that all control rods were fully inserted.	Operator tripped B and C reactor feed pumps (RFP), B and C condensate booster pumps, and C condensate pump. Reduced loading on reactor feed pump subpanel to prevent "over shoot" on reactor level return. RCIC started manually as backup.	Pumps responded to trip signal and reactor level was maintained by reactor feed pump A and RCIC.
~ 1254	Unit conditions indicated need for tripping turbogenerator.	Assistant shift engineer (ASE) initiated turbine trip upon observing generator load at 100 MW. Also opened generator field breaker and motor-operated disconnects (MOD's).	Turbine bypass valves opened to compensate for turbine valve closure and maintain pressure normal. The main-steam isolation valves (MSIV's remain open allowing reactor pressure control through the turbine bypass valves to main condenser heat sink.
		Unit operator inserted source range and intermediate range neutron monitors and observed reactor power decrease.	Neutron monitoring responded normally.
	HPCI started.		HPCI automatically aligned in normal injection mode to reactor vessel.
	Reactor water level restored to approximately normal range.	Operators shut down HPCI and RCIC.	HPCI and RCIC shutdown. Problems incurred upon shutdown with valve operation associated with th systems.
~ 1255	Lost 120-V unit preferred power. One of the feeds from this source is the unit control rod position indication on panel 9-5 (reactor control panel).	Operator placed reactor mode switch in "Refuel" mode to verify one rod withdraw permit. (All rods must be fully inserted or the indicating light for one rod withdraw in refuel mode will not illuminate.)	Received white permit light.
	Lost all neutron monitoring.	Operator observed no indication on average power range, intermediate range, or source range monitor.	Capability to monitor core was lost.

Time	Event	Action	Response or Nonresponse
~ 1256	By this time the following electrical boards were lost: 1A 250-V D.C. Reactor MOV board* 1B 250-V D.C. Reactor MOV board 1A 480-V A.C. Reactor MOV board 1B 480-V A.C. Reactor MOV board 1C 480-V A.C. Reactor MOV board 1A 480-V A.C. Shutdown (SD) board 1B 480-V A.C. Shutdown board 120-V A.C. unit preferred	Indication from the unit control room as to electrical sources feeding the various equipment and as verified by ASE as he checked the individual boards.	This caused the loss of vital equipment being fed from these electrical boards. Loss of power to MSIV's caused them to go closed (all 4 outboard valves), placing the unit in isolation from the main condenser heat sink and cutting off the steam supply to the reactor feed pump turbines. All emergency core cooling systems were lost with the exception of 4 relief valves which could be operated from the unit control board.
~ 1258	Reactor pressure rapidly increased to 1,100 psig.	ASE was unsuccessful in opening MSIV's from backup control center. Operator manually opened main-steam relief valves; then closed as pressure came back to desired range. Attempts to place RCIC in service were unsuccessful from control room or backup control panel.	Relief valves opening and closing to maintain pressure between 1,080 and 1,100 psig. Relieving to the suppression pool (torus). Pressure decreased to 850 psig; then rapidly increased to 1,080 psig. Valve 71-2 (steam supply to turbine) was apparently the only valve lost on RCIC but rendered it inoperable...This valve was later opened by use of temporary power. The HPCI was previously rendered inoperable by loss of valve controls.
~ 1259	Reactor water level decreasing due to almost constant blowing down to the torus. Torus cooling became essential.	The only water input left with the capability to overcome a pressure above ~ 350 psig was the control rod drive pump; it was increased to the maximum. ASE was unsuccessful in placing emergency power on RHR valves at local MOV board. (Those valves required for torus cooling.) Shift engineer and two electricians making attempts to restore 480-V 1A and 1B reactor MOV boards and 250-V D.C. boards.	RHR system was unavailable for torus cooling as a result of electrical board losses.
1300	4-kV SD board C undervoltage shutdown bus 2 undervoltage. (As noted on electrical printer.)		4-kV voltage continued to be supplied to SD boards A and B by shutdown bus 1. Shutdown boards C and D transferred to D/G's C and D.

*MOV - Motor operated valve

Time	Event	Action	Response or Nonresponse
1320	SD bus 1 undervoltage.		A and B D/G's tied onto their respective boards. The A and B diesel generators were running and up to speed prior to this but had not received an undervoltage signal to close them onto their boards until this event. This placed all the 4-kV shutdown board equipment upon the diesel generators.
1321	Process computer lost.		No further printout until ~ 1600 hours.
~ 1330	Decision made to depressurize reactor by blowing down to torus.	Manually opened 4 main-steam line relief valves. Checked condensate booster pumps running. Reactor feed pump bypass valve 3-53 checked opened ready to admit water from condensate system.	Reactor pressure decreased; water level decreased Reactor water level dropped from normal 201 inches above top of active fuel to 48 inches abo top of active fuel.
~ 1334	Shutdown bus 2 transferred. (In maintaining a normal configuration on the plant electrical system, the 4-kV shutdown boards will be lined up to feed from the unit or offsite source feed. This allows the diesel generators (D/G's) to be a highly reliable backup, giving tw sources of voltage should the need arise.)	Manually initiated by ASE by normal procedure of synchronizing the D/G's with the SD bus; then dropping off D/G feed to the SD board.	Bus 2 energized from unit 2. A,B, and D SD boards transferred to SD bus 2. D/G's remained on running standby.
	Shutdown bus 1 continued to be deenergized.	ASE was unsuccessful in an attempt to manually energize.	
	Shutdown board C remains energized from C D/G.	ASE was unsuccessful in an attempt to manually change C SD board feed from D/G to shutdown bus 2.	Breaker stayed closed for 5 to 10 seconds; then opened. Feed transferred back to D/G.
	D shutdown board deenergized for ~ 5 minutes.	ASE reenergized D board.	D board remained feeding from shutdown bus 2.
~ 1345	Restored unit preferred from unit 2.	ASE manually transferred.	Unit preferred back on both units.
	Reactor steam pressure decreased to 350 psig.	From continued manual operating of relief valves.	Reactor water level increasing as a result of condensate booster pump input.
~ 1355	Water level approaching normal.	Attempted to throttle the feedwater bypass valve 3-53.	No response on feedwater bypass valve 3-53.

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Time	Event	Action	Response or Nonresponse
~ 1357	Water level going high.	AUO dispatched to bypass valve. Manually closed down on valve.	Restored level to approximately normal.
	Outgoing (PAX) phones and page inoperable for some time.	All operations requiring control room monitoring set up on a periodic in-call basis.	
~ 1400	Reactor steam pressure at 200 psig.	Operator maintaining reactor steam pressure through four relief valves and level control through RFP bypass and CRD pumps. This had to be controlled via phone communication since the paging system was inoperative.	
	480-V SD boards A and B restored.	ASE manually initiated.	Boards appeared heavily loaded as indicated by loud "humming." Boards remained in service.
		ASE tried to restore reactor 480 MOV boards A and B and reactor 250 MOV boards A and B.	Initially unsuccessful.
	Off-duty maintenance personnel began reporting.	Electricians and operators working to restore these electrical boards by isolating faulted circuits.	Restored approximately two hours later.
~ 1448	Voltage lost to 4-kV shutdown board C.		There was no control room indication of this condition.
	B D/G found tripped with field breaker open.	ASE closed field breaker on B D/G and brought back to running standby.	
~ 1500	Attempt made to align one RHR system up for torus cooling and the other for SD cooling.	Four AUO's working in pairs using breathing air packs. Made two entries, but insufficient air supply aborted attempts.	
~ 1527	Voltage restored to C shutdown board.	ASE found C D/G running at approximately 1/2 speed. Brought D/G to synchronous speed and closed breaker to board.	C shutdown board was deenergized. C shutdown board was also lost from 1545 to 1557. However, at ~ 1630 C D/G was tied onto the board, its breaker tripped, and prevented C D/G from being used.
~ 1600	RHR system 1 aligned for torus cooling.	Decision made not to start in this condition since it could not be established that system was charged with water.	This system was subsequently checked for proper alignment and charge and placed in service later.

Time	Event	Action	Response or Nonresponse
~ 1630	480-V reactor MOV board 1A reenergized.	Result of electrician and operator action.	Energized electrical board allowed main turbine to be placed on turning gear and reactor protection system MG set A to be placed in service which energized trip channel A. Restored power to 1/2 of the process monitoring. Restored power to those ECCS valves feeding from that board, etc.
	Valve restoration to ECCS equipment.	By operator interview, it has been established that the following procedure was used in valve operation where valves were not operable from the control room. Placed select switch in emergency position at the electrical board, run valve to desired position, reopened breaker, immediately returned to control room and established that that level was not affected by possible electrical fault misalignment. Tagged valve control switch on unit control board showing valve position. This was a safeguard against draining the vessel down. ASE observation of conditions as appeared on panel 9-3 in control room.	Level remained normal. Core spray loop I A and C pumps appear operable from unit control board. All valves and both pumps had indicating lights. Core spray loop II had a few valves that were inoperable. RHR loop II had a few valves available. RHR loop I--same.
~ 1640	Request to start reactor building exhaust fan to remove smoke and fumes.	Started locally from 480-V reactor building vent board.	Fan responded normally. Dampers controlled manually at the damper.
~ 1700	Request to stop reactor building exhaust fan as airflow appeared to aide fire.	Stopped locally by operator.	Fan stopped.
~ 1800	Relief valves inoperable by remote manual control from benchboard due to loss of instrument and control (I&C) voltage to solenoid in air supply to diaphragm valve in air header to primary containment.	Operator observed lights indicated relief valves open. Other indications suggested that valves were closed. Restarted drywell air compressor. Craftsman bypassed solenoid valve to provide control air supply to primary containment equipment.	Reactor pressure increasing from 800 psi. The compressor started but discharge isolation prevented airflow to primary containment and relief valve control. Allow relief valve remote manual operation at 2130 hours.

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Time	Event	Action	Response or Nonresponse
~ 1900	Decision made to restart the reactor building exhaust fan.	Manually initiated.	Remained in service.
	PAX telephones restored to unit control room.	ASE cleared problem on phones.	Gave control room the capability to call out.
~ 2000	High torus level from earlier blowdown.	Manually aligned and started RHR drain pump to main condenser hotwell.	Torus level decreased. Hotwell level increased.
~ 2040	Venting drywell via standby gas treatment system to plant stack. Pressure was about 2.5 psig.	Steamfitters manually opened 2-inch vent to standby gas treatment system.	Drywell pressure decreasing.
~ 2150	Relief valves operable by remote manual control.	Switch left in open position anticipating voltage return. Manually operated relief valves to reduce reactor pressure.	Reactor pressure decreasing from 550 psig maximum.
~ 2200	Secondary containment reestablished.	Operator stationed at reactor building entrance.	Shift engineer approval before entering required. Breathing apparatus required.
~ 2230	ASE made attempt to prove D D/G operable from electrical control board in the control room.		Could not be operated from control room.
		ASE synchronized to D 4-kV shutdown board, locally, picked up load, placed on standby.	Could be used if needed by operating from D shutdown board.
3/23/75 ~ 0000	Need for flushing RHR system II prior to placing in shutdown cooling.	Existing procedures could not be used under present circumstance. Two senior reactor operators approved temporary flushing procedure.	System flushed and placed in service at 0410.
~ 0100	Two source range monitors placed in temporary service located on the reactor side of the fire.	Licensed reactor operator stationed at these monitors in the area of unit 1 drywell continuous air monitor unit.	Established capability to monitor core. 10 counts per second reading on monitors.
~ 0130	Torus cooling continues to be a necessity as blowdown continues.	Valves aligned manually by operators and system placed in service.	Decreasing torus temperature.
~ 0212	Torus level instrumentation in service.		Level indicated +1". (Normal level is indicated as 0 with a deviation of + or - 5".)
~ 0245	Restoration of equipment had progressed to the point that A and C core spray pumps could be tested from panel 9-3 in unit control room.	Operator action from unit control room.	Pumps and injection valves operable, thus giving part of the ECCS equipment available if needed.
~ 0410	Shutdown cooling achieved by normal flow path.	Manually aligned system.	Allowed operator control of vessel temperature.

BROWNS FERRY UNIT 2

SEQUENCE OF
SIGNIFICANT OPERATIONAL EVENTS
AT TIME OF FIRE

Time	Event	Action	Response or Nonresponse
3/22/75 Prior to 1300	Initial Condition	Routine Operation	Unit Load 1,100 MWe
~ 1300	4-kV Shutdown bus 2 deenergized (relay action).		Lost reactor protection system (RPS) motor generator (MG) set 2B; 1/2 scram on RPS giving red lights on panel 9-5; reactor recirculation pump automatically decreasing reactor power. Lost voltage to instrument and control bus B. Lost indicating lights on system I residual heat removal (RHR) and system I core spray - alarms on RHR and core spray "start," "overcurrent," "pump trip."
	Operator observed decreasing reactor power indication and many scram alarms on control panel.	Operator placed reactor mode switch in "shutdown" and inserted nuclear instrumentation (source and intermediate range).	Reactor scrammed inserting all control rods.
~ 1301	Reactor water level dropped and returned to normal (normal reaction from trip).	Tripped reactor feed pumps A, B, and C - Tripped turbine. Tripped exciter field breaker and opened generator motor-operated disconnects (MOD's).	Equipment response normal.
1308	Main-steam isolation valves (MSIV) closed.	Operator initiated reactor core isolation cooling (RCIC) for level control; initiated high pressure cooling injection (HPCI) for heat sink. Manually initiated relief valves for pressure control.	Equipment response normal. After this start and before ~ 1415 RCIC and HPCI tripped several times from high reactor water level. Neither of these could be restarted with the controller in "manual." Operator was unable to get any signal from the subpanel control in "manual." Pumps would start with controller in "automatic." At ~ 1345 HPCI was restarted and brought to ~ 1/4 speed. It held for about 1 minute. The speed then dropped off with no further response from HPCI; thereafter it was unavailable.

Time	Event	Action	Response or Consequence
~ 1320	Lost remote manual operability of relief valves depriving operator of ability to reduce reactor pressure below set point.	Unit 2 assistant shift engineer made an attempt to operate relief valves from backup control panel but was unsuccessful.	Relief valves continued to lift on pressure. Maintaining reactor pressure at 1080 psig and below.
	Torus temperature increasing due to relief valve discharge into torus.	Placed the following pumps in service to establish torus cooling: B RHR pump, B2 RHR service water pump, and B1 emergency equipment cooling water pump.	Torus cooling established at ~ 1330. Torus temperature did not exceed 135° F.
~ 1400	Reactor depressurizing apparently from a relief valve that had lifted on pressure and stuck open.	No indication of coolant leak, and pressure decreasing at desired rate.	Reactor pressure decreasing at desired rate: 150 psig at 1500 65 psig at 1900 10 psig at 2040
~ 1415	Pressure starting to decrease.	Placing the condensate system in service.	Anticipating the pressure level that reactor vessel could be supplied from that source.
	Remote manual operability of relief valve restored.	Maintenance and operations personnel working during this period of non-operation checking the instrument and control (I&C) voltage to solenoids, drywell air compressor for proper operation and cutting in the backup control air supply. It is uncertain which of these operations reestablished remote operability of relief valves.	Gave operator discretion on relief valve operation.
~ 1430	Loss of some reactor water level instrumentation.	Determined that level indicator 36E appeared to be reliable and that 2 Torrey level indicators in backup control center corresponded with this indication.	Reactor water level never decreased below 160" above the top of the active fuel. Other level indication began to respond at ~ 1450.
~ 1450	Torus level increasing due to relief valve discharge.	Manually aligned RHR drain pump to transfer torus water to condenser hotwell.	Torus level never increased above + 5".
~ 1527	Voltage restored to C shutdown (SD) board.	Restored power to 480-V SD board 2A by manual operator action. Started B2 RHR service water pump on. Started B RHR pump on. Placed turbine on turning gear (T.G.)	These power supplies allowed turbine to be placed on T.G. and B RHR pump to be tested.
~ 2010	Condenser heat sink available. Allowed use of turbine bypass valves to reduce pressure.	Cleared up electrical trouble with mechanical vacuum pumps and established vacuum in main condenser.	Vacuum above 7" Hg allowing opening of turbine bypass valves for steam admission.
~ 2020	Torus temperature within limits; shut down torus cooling to allow flushing of lines for reactor shutdown cooling.	Aligned valves and flushed system II prior to placing system II in reactor shutdown cooling mode.	
~ 2045	Reactor pressure at 10 psig.	Opened reactor headvents.	
~ 2240	Reactor in shutdown cooling using RHR system II.	Manually aligned system.	Shutdown cooling achieved by normal flow path.

UNITED STATES GOVERNMENT

Memorandum

TENNESSEE VALLEY AUTHORITY

TO : Preliminary Investigation Committee for Fire at Browns Ferry Nuclear Plant

FROM : James E. Watson, Manager of Power, 818 PRB-C

DATE : March 31, 1975

SUBJECT: ESTABLISHMENT OF COMMITTEE TO INVESTIGATE THE MARCH 22, 1975, FIRE AT BROWNS FERRY

This memorandum will confirm and formalize the establishment on March 23, 1975, of a preliminary committee to investigate the fire at Browns Ferry on March 22, 1975. The committee reported to the site on March 24, 1975. It is composed of the following members:

H. S. Fox, Chairman	- Division of Power Production
M. W. Sprouse	- Division of Engineering Design
Charles Bonine	- Division of Construction
David G. Powell	- Division of Law
Felix A. Szczepanski	- Power Manager's Office, Safety Review Board Staff
Harry S. Collins	- Secretary, Safety Review Board

Your participation on this committee is greatly appreciated and while we recognize that it will in all probability cause you personal hardships, we believe it is extremely important to give the committee your full effort for the immediate future.

Attached is an outline of the committee's assignment. Many of the points covered in the outline have already been addressed but this will confirm the assignment to the committee.

Again, I appreciate your serving on this committee and if I can help you in any way, please let me know.

Attachment
CC: R. M. Davidson, 303 PRB-C
J. E. Gilleland, 831 PRL-C
G. H. Kimmons, 607 UB-K
R. H. Marquis, 629 NSB-K
Nuclear Safety Review Board, 210 PRB-C
H. G. Parris, 403 PRB-C
E. P. Thomas, 716 EB-C

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PRELIMINARY INVESTIGATION

of

BROWNS FERRY FIRE

Preliminary Incident Evaluation Committee

Scope

Conduct an early fact-finding investigation into events leading up to, during and after the incident until plant conditions were stabilized. Consider actions of people involved, applicable administrative controls, response and nonresponse of plant systems, alternative measures taken, utilization and adequacy of plant firefighting equipment, assessment of extent of damage, status of plant and plant systems; determine if there were any radioactive releases, radioactive exposure and/or injury to plant personnel, and if so, determine the extent thereof; ensure preservation of adequate incident documentation; and provide a point of reference for other evaluations.

Other Considerations

The committee should:

- Use its discretion in extending its scope and in carrying out its functions to achieve its objectives.
- Recommend and seek approval for additional committee members (i.e., consultants or other TVA members); consider qualifications.
- Make preliminary report to the Manager of Power within two weeks (by April 7).

Membership

Harry Fox (Chairman)	- DPP
F. A. Szczepanski	- NSRB
C. Bonine	- DEC
M. Sprouse	- DED
D. Powell	- Law
H. S. Collins	- NSRB Staff

APPENDIX B

KEY PHOTOGRAPHS OF FIRE AREA

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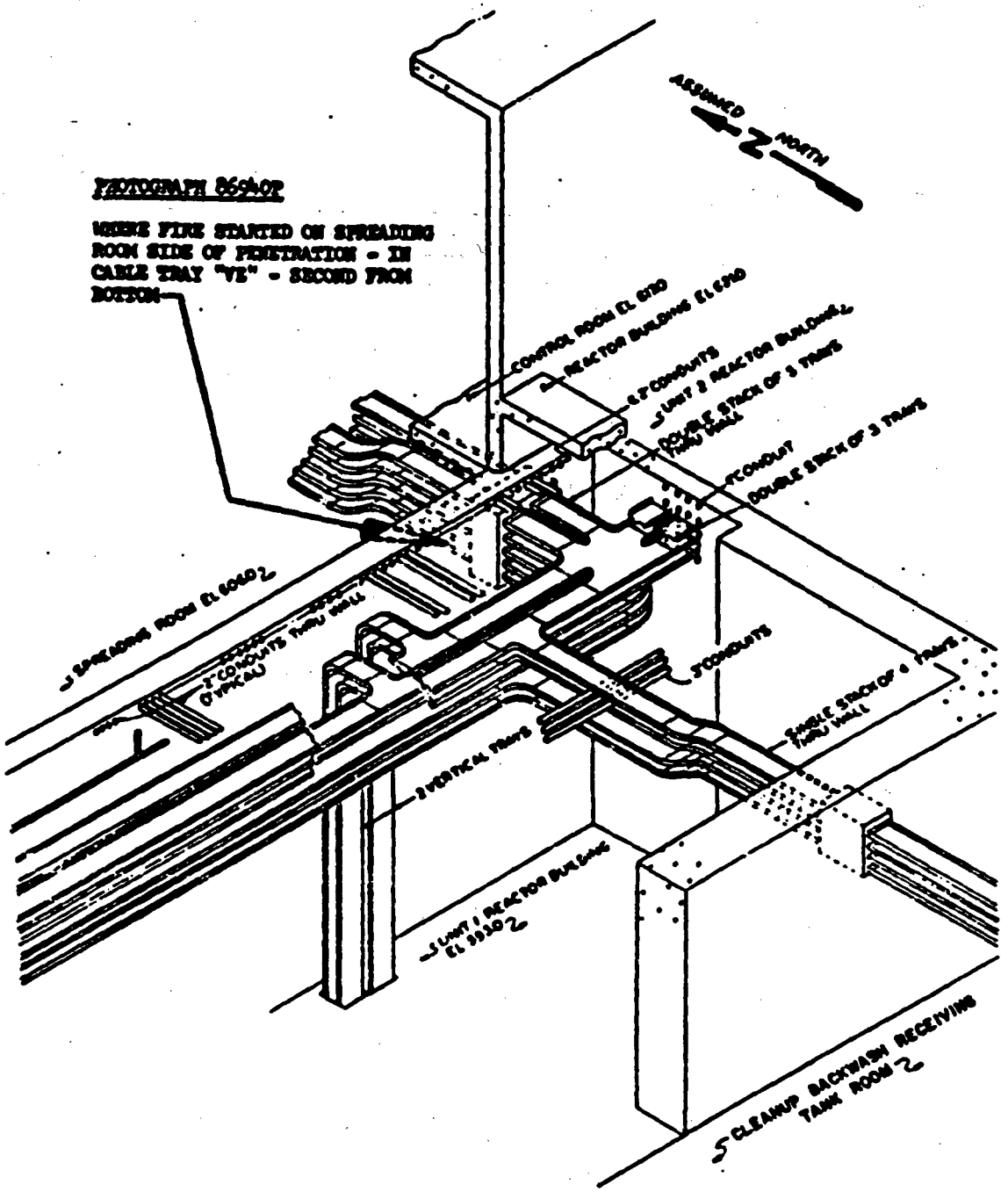
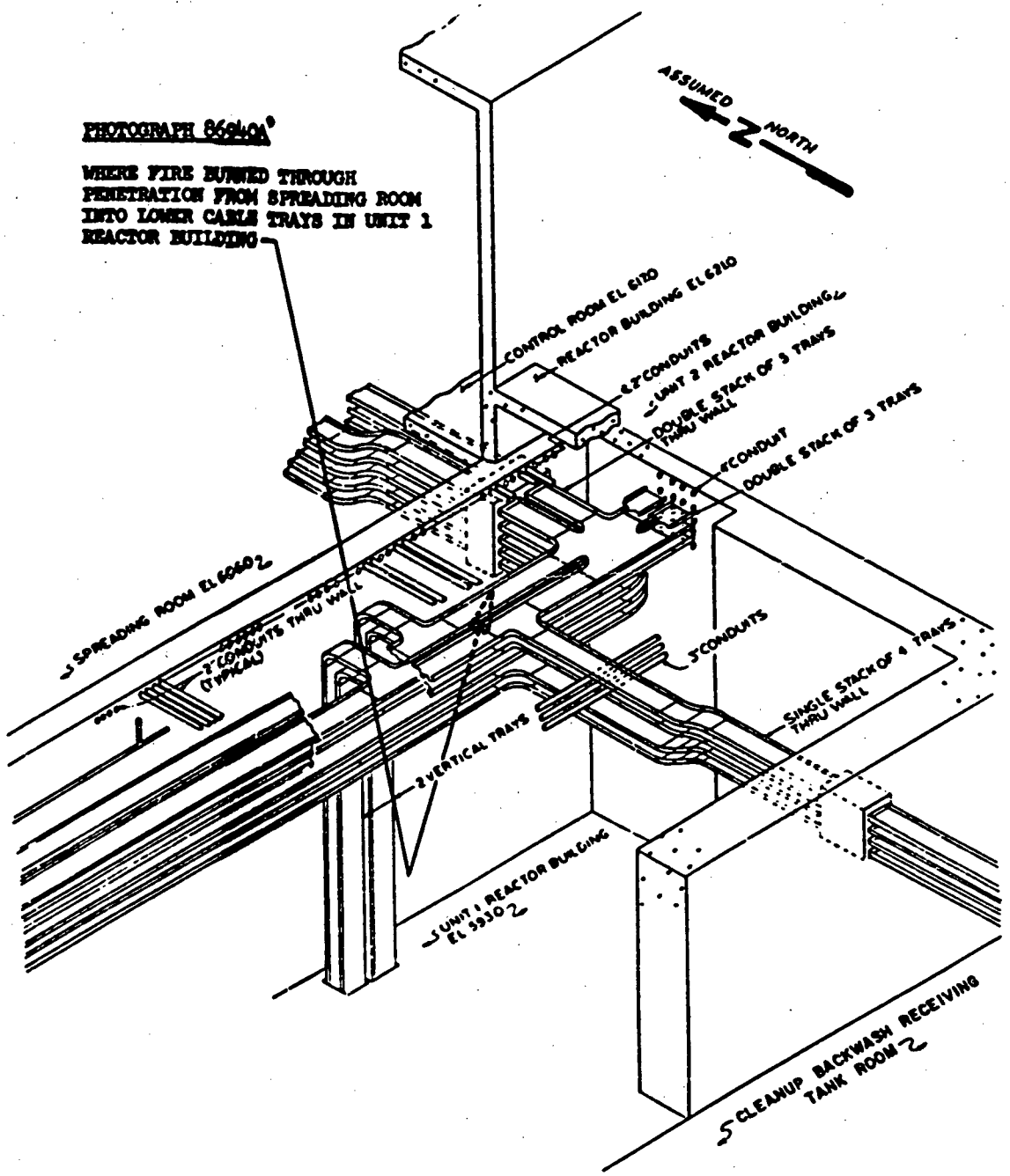
1. Photograph 86940P - Penetration, Spreading Room Side
2. Photograph 86940A - Penetration, Reactor Building Side
3. Photograph 67P1991 - General View, Northeast Corner of Reactor Building
4. Photograph 86940H - Horizontal Trays in Reactor Building, Westward
5. Photograph 86575P - Horizontal Trays in Reactor Building, Eastward
6. Photograph 86575N - Horizontal Trays in Reactor Building at Intersection
7. Photograph 86940I - Horizontal Trays in Reactor Building, Southward
8. Photograph WH-K-86577-B - Penetration at South Wall
9. Photograph 89438K - Conduit Damage
10. Photograph WH-K-86577-C - Penetration, Reactor Cleanup Tank Side



AREA OF FIRE

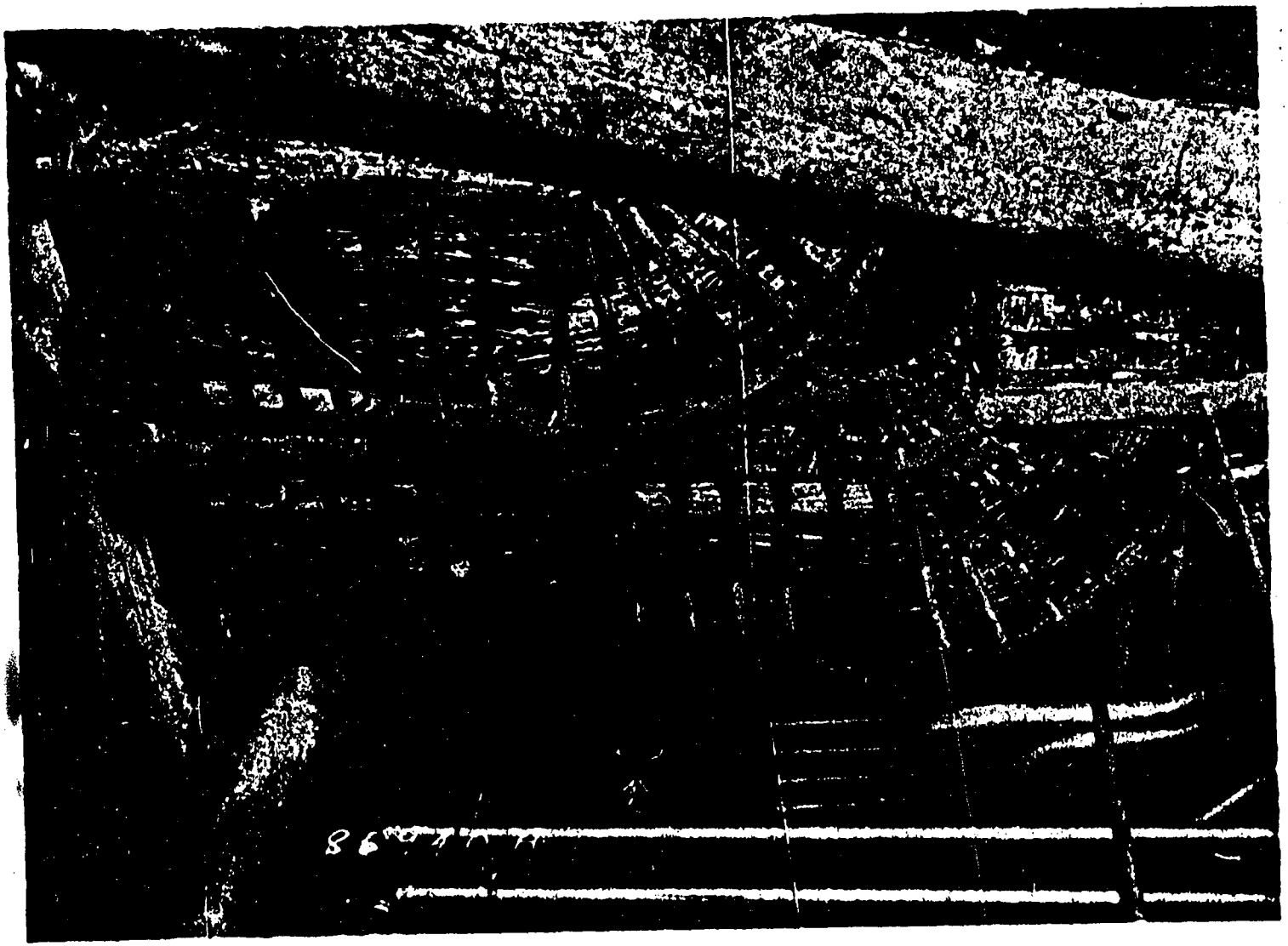
1

86940P

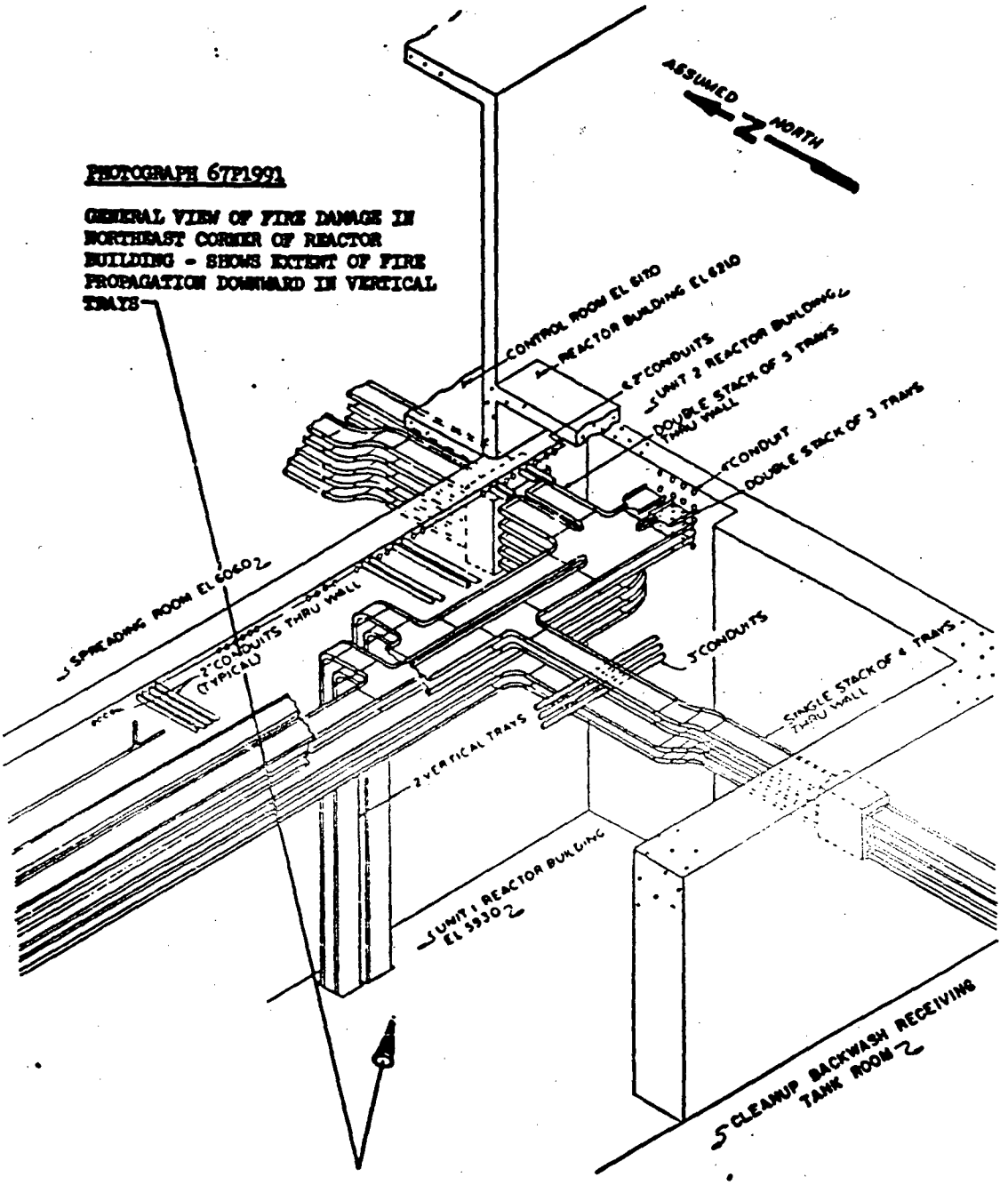


AREA OF FIRE

2

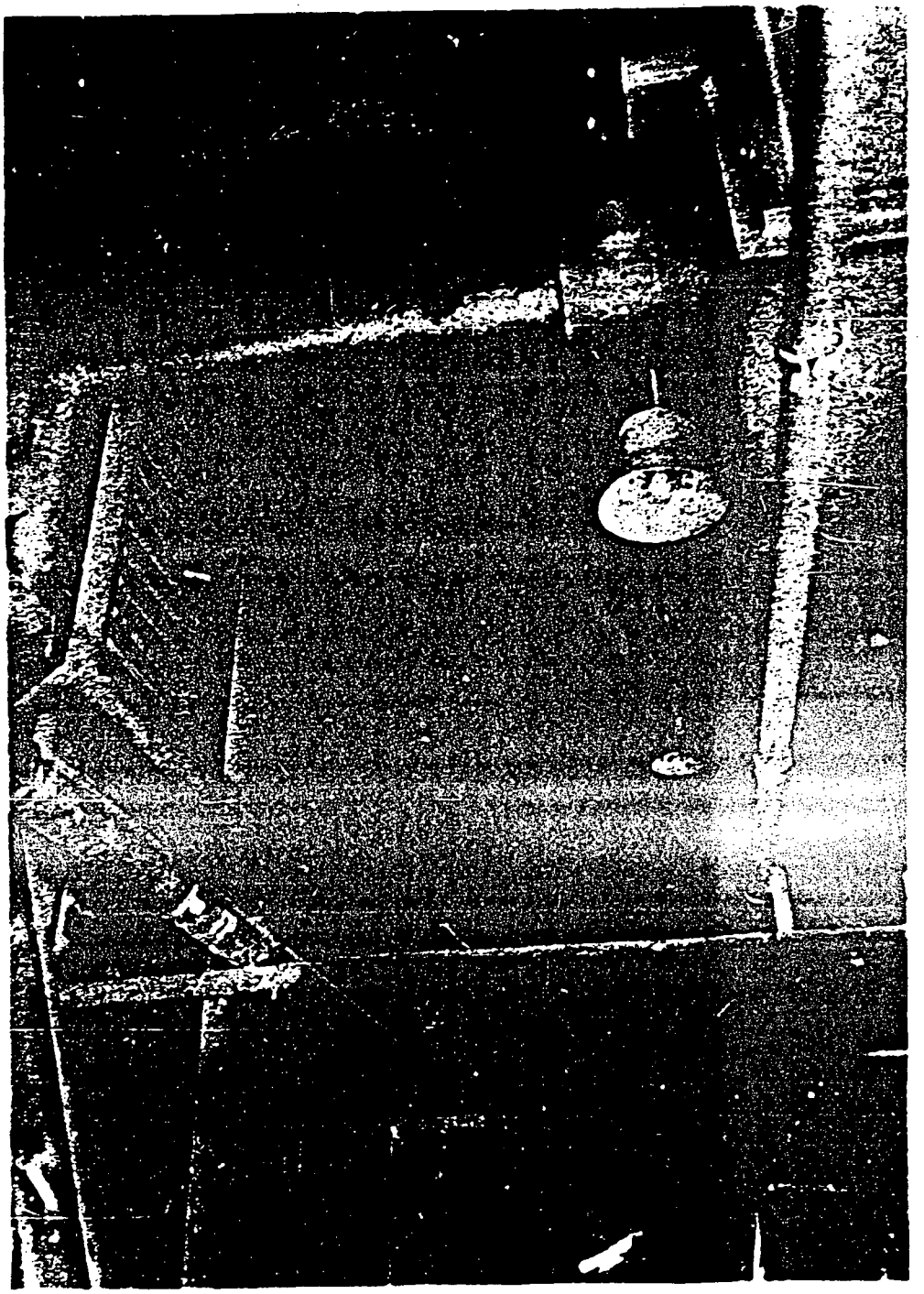


86940A



AREA OF FIRE

3



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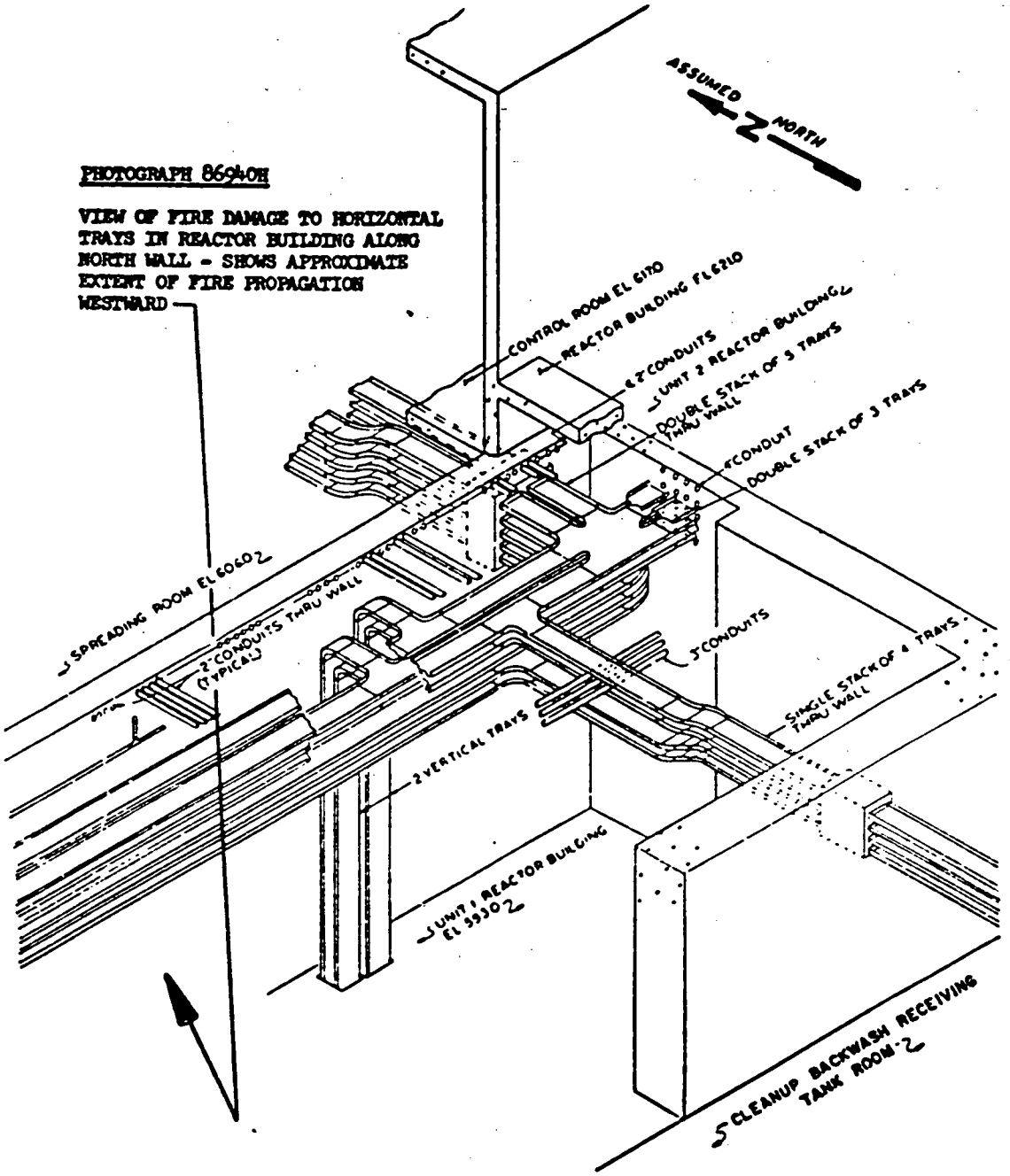
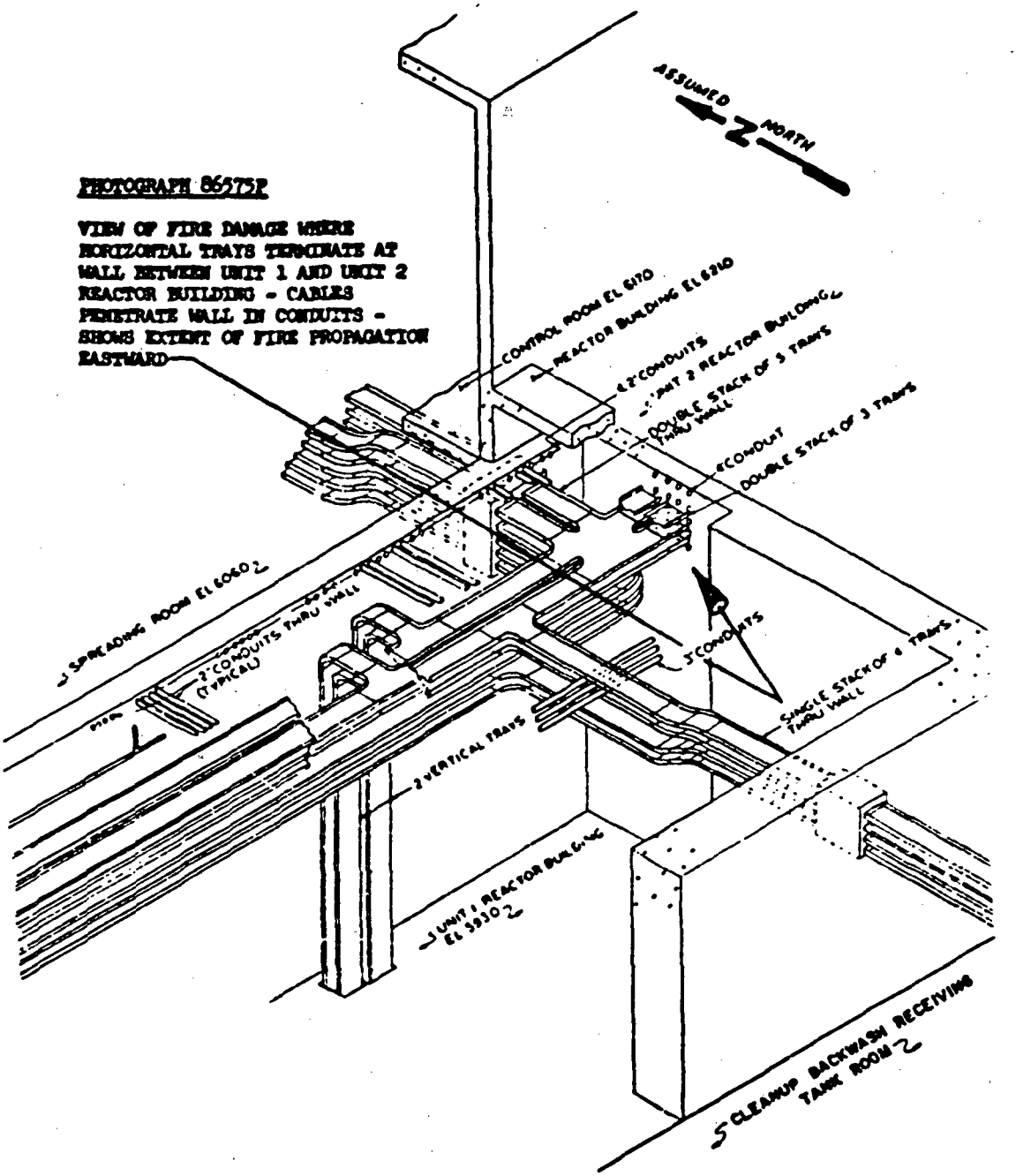

ASSUMED NORTH
PHOTOGRAPH 86940H
VIEW OF FIRE DAMAGE TO HORIZONTAL TRAYS IN REACTOR BUILDING ALONG NORTH WALL - SHOWS APPROXIMATE EXTENT OF FIRE PROPAGATION WESTWARD
CONTROL ROOM EL 610
REACTOR BUILDING EL 620
& 2" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
1" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 606
2" CONDUITS THRU WALL
2" CONDUITS (TYPICAL)
5" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 593
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE

H0469 8



AREA OF FIRE

5

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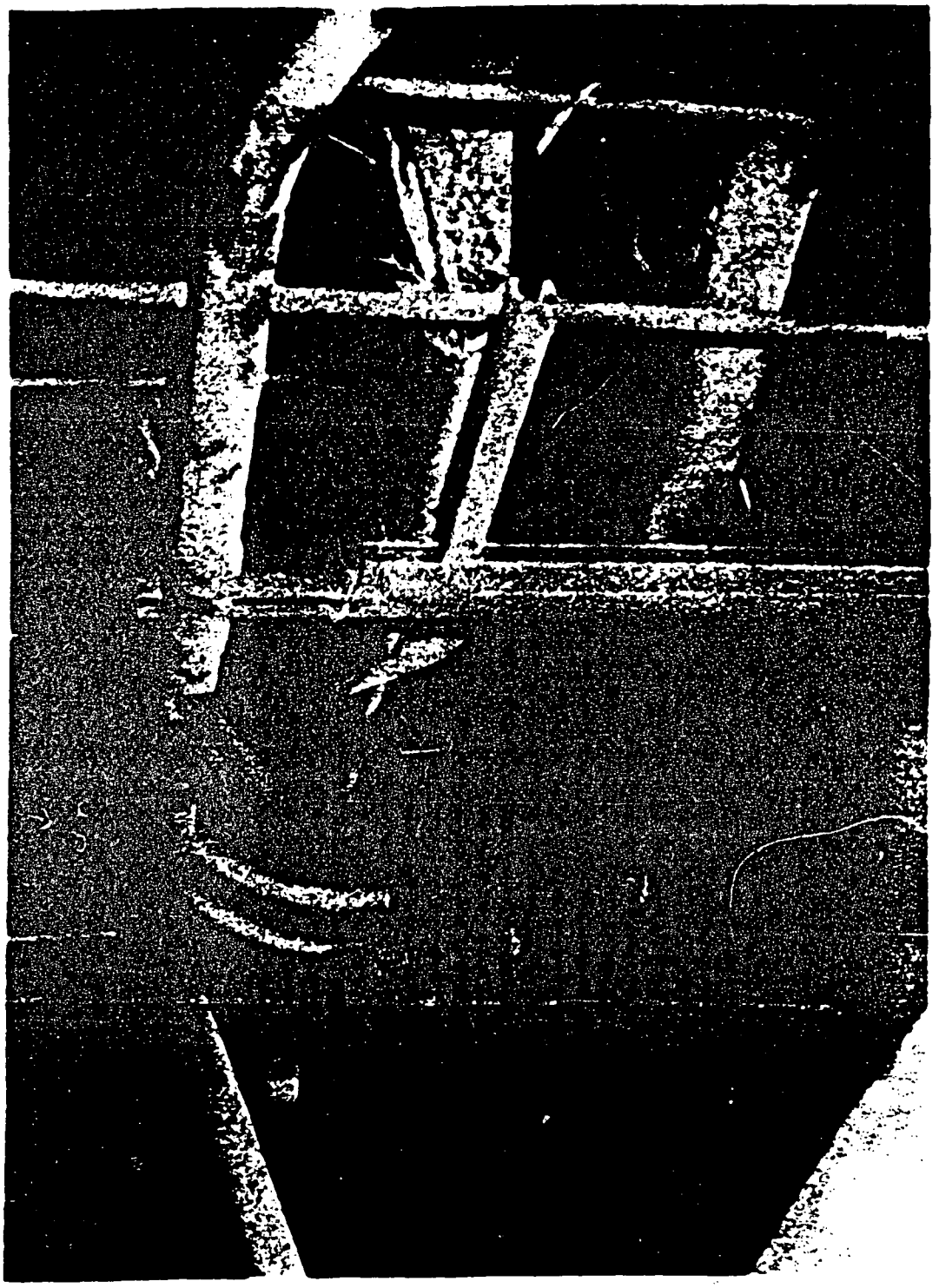

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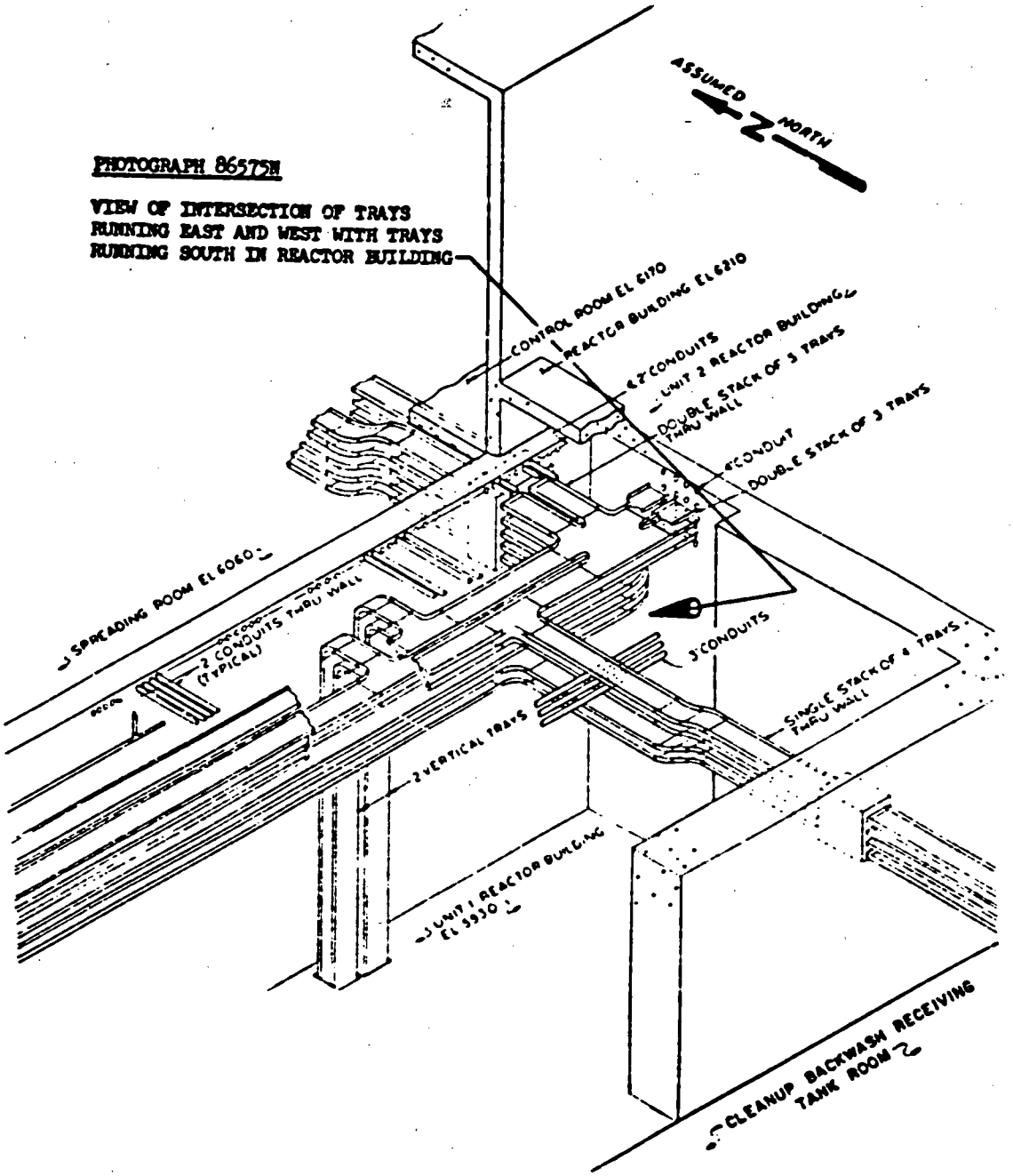
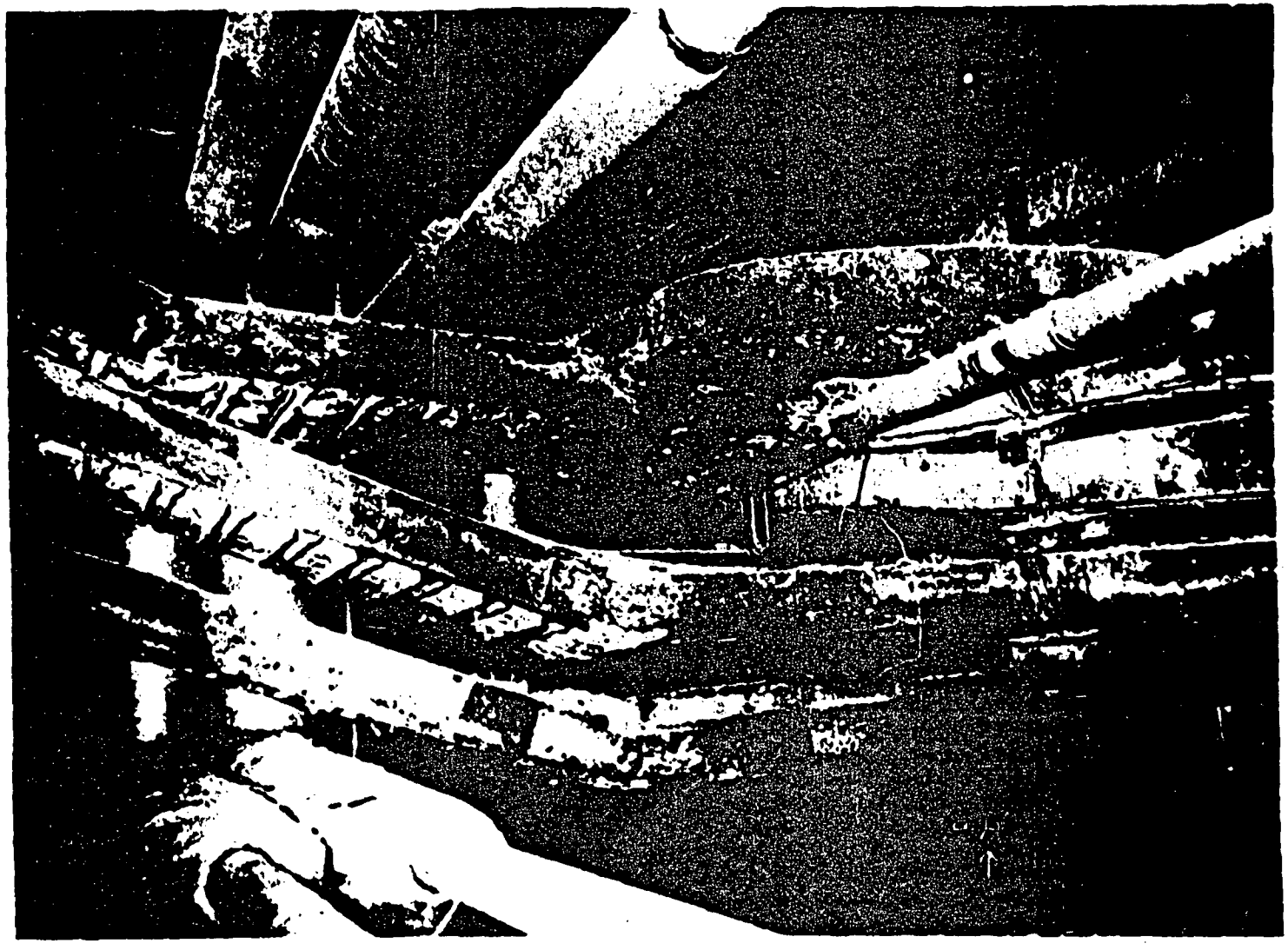

PHOTOGRAPH 86575H
VIEW OF INTERSECTION OF TRAYS
RUNNING EAST AND WEST WITH TRAYS
RUNNING SOUTH IN REACTOR BUILDING
ASSUMED NORTH
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
4" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS
THRU WALL
4" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2 CONDUITS THRU WALL
(TYPICAL)
3" CONDUITS
SINGLE STACK OF 4 TRAYS
THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING
EL 5950
CLEANUP BACKWASH RECEIVING
TANK ROOM

AREA OF FIRE



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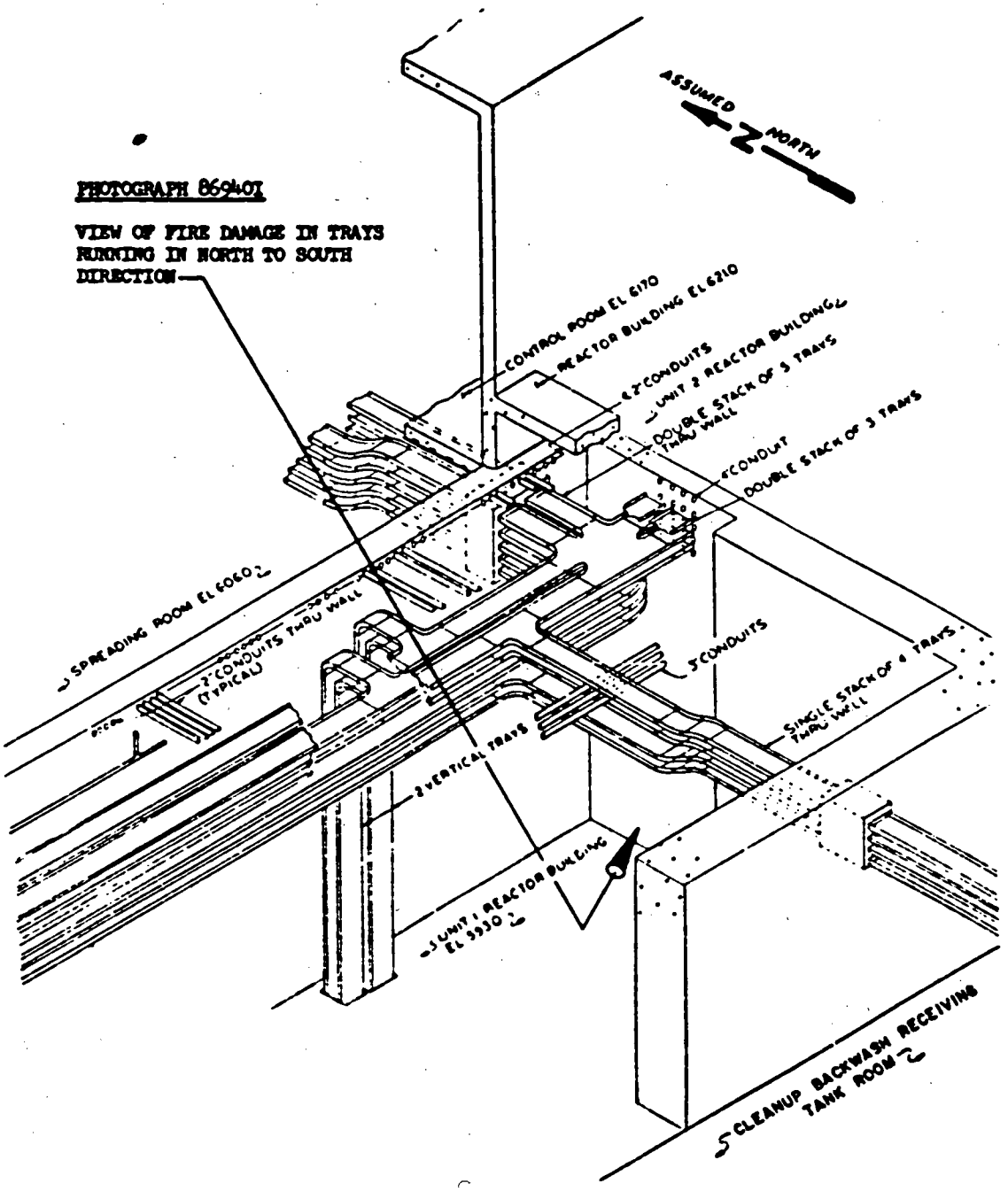
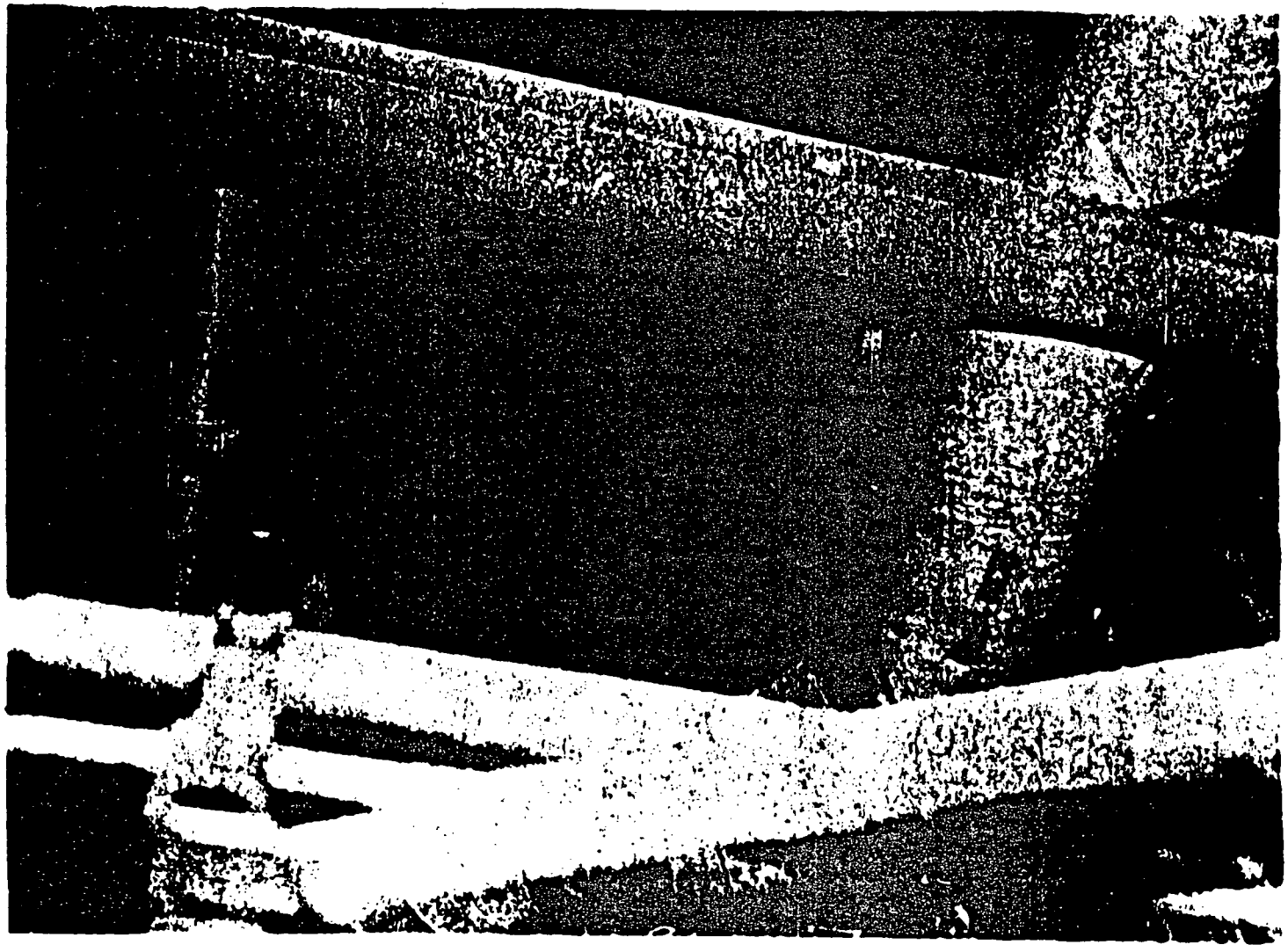

ASSUMED NORTH
PHOTOGRAPH 869401
VIEW OF FIRE DAMAGE IN TRAYS RUNNING IN NORTH TO SOUTH DIRECTION
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
4" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
4" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL (TYPICAL)
5" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 3930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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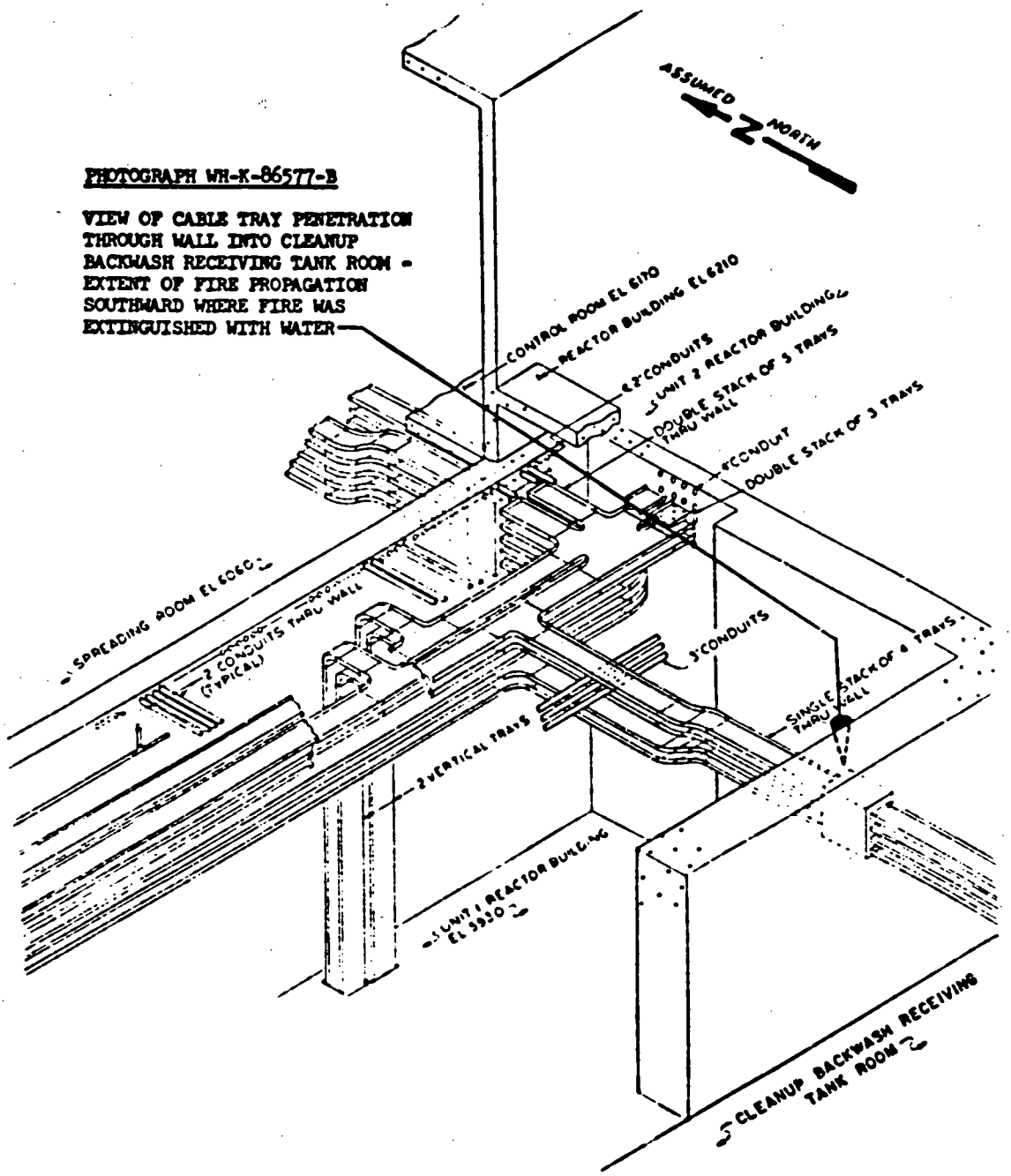
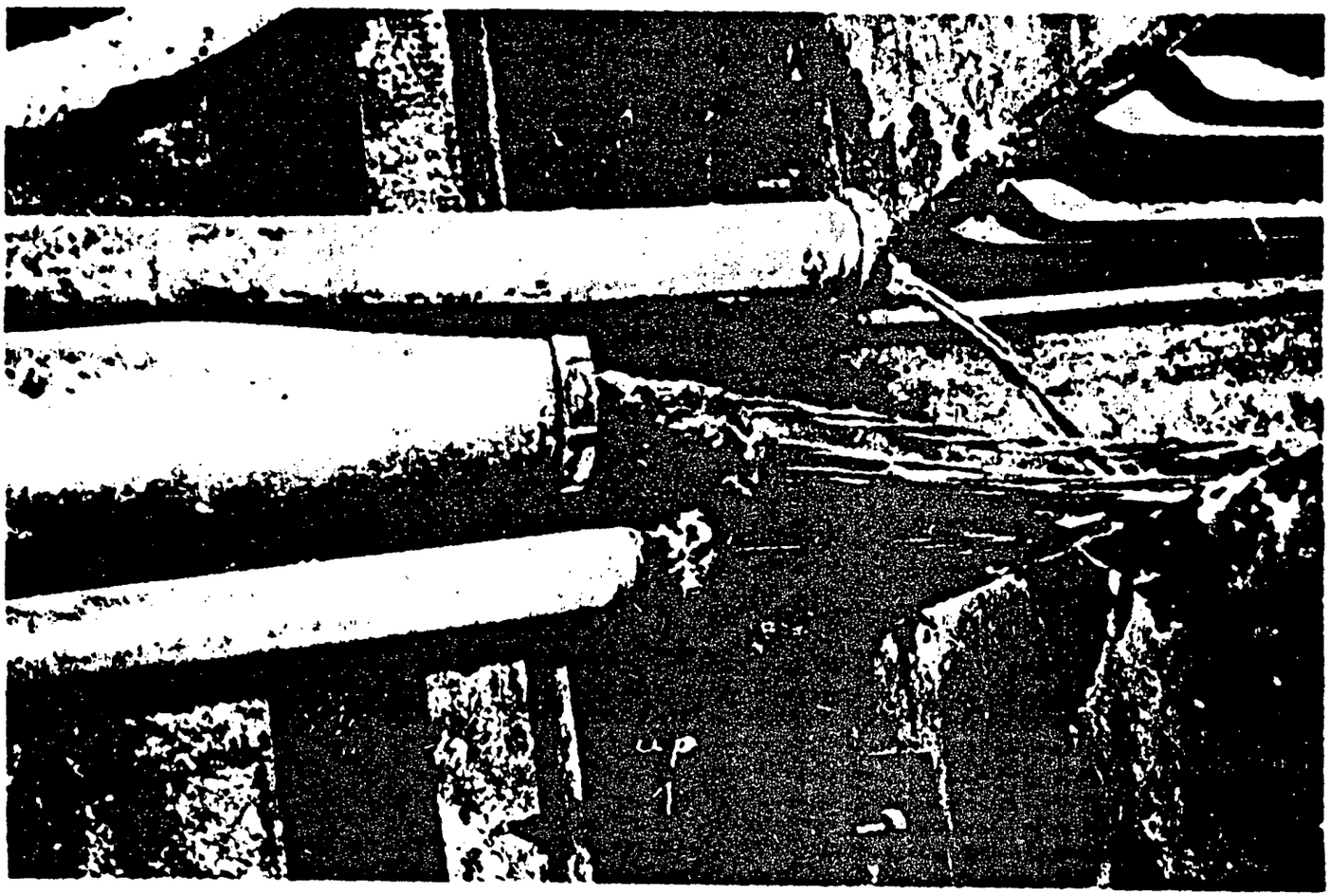


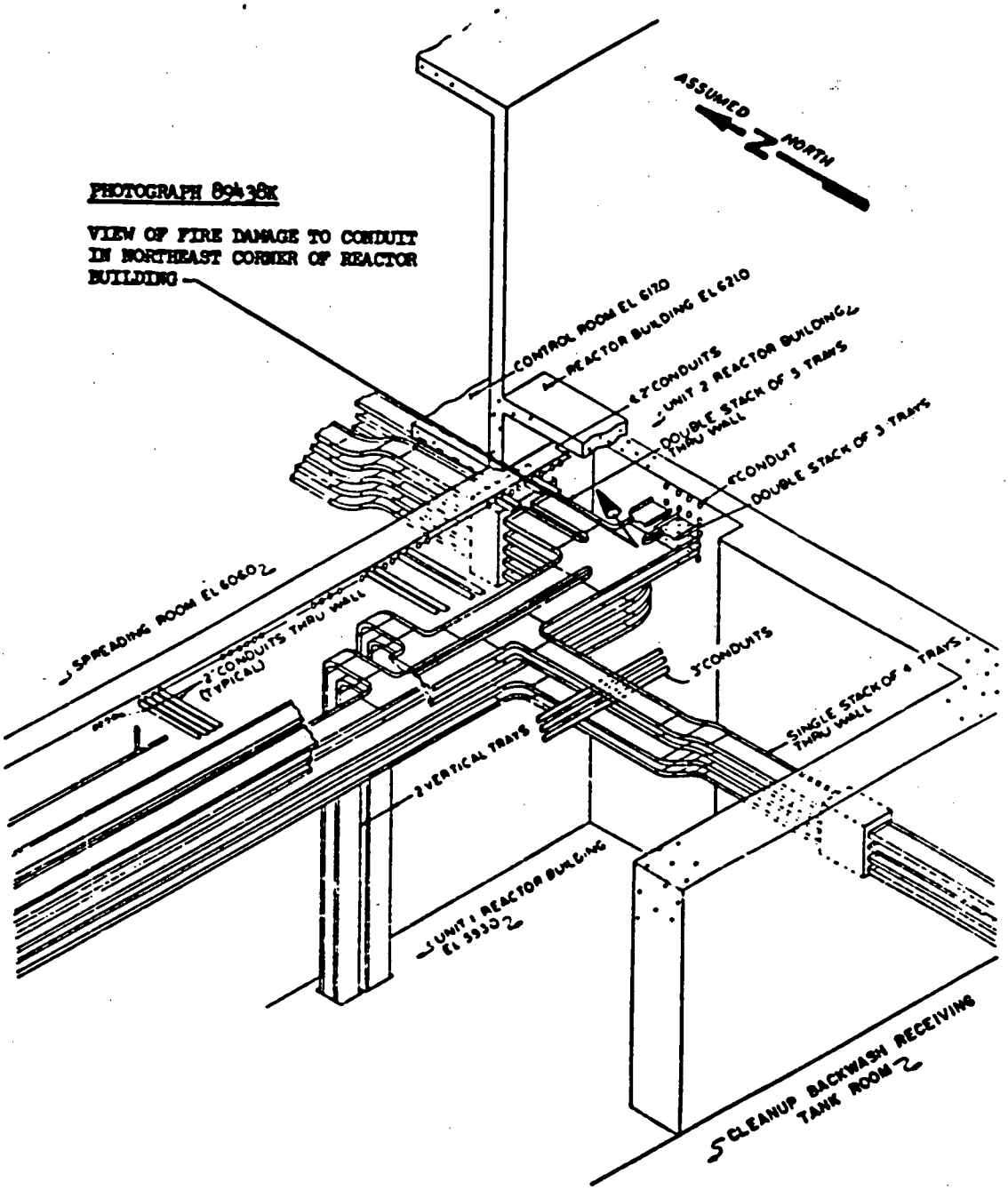
PHOTOGRAPH WH-K-86577-B

VIEW OF CABLE TRAY PENETRATION THROUGH WALL INTO CLEANUP BACKWASH RECEIVING TANK ROOM - EXTENT OF FIRE PROPAGATION SOUTHWARD WHERE FIRE WAS EXTINGUISHED WITH WATER

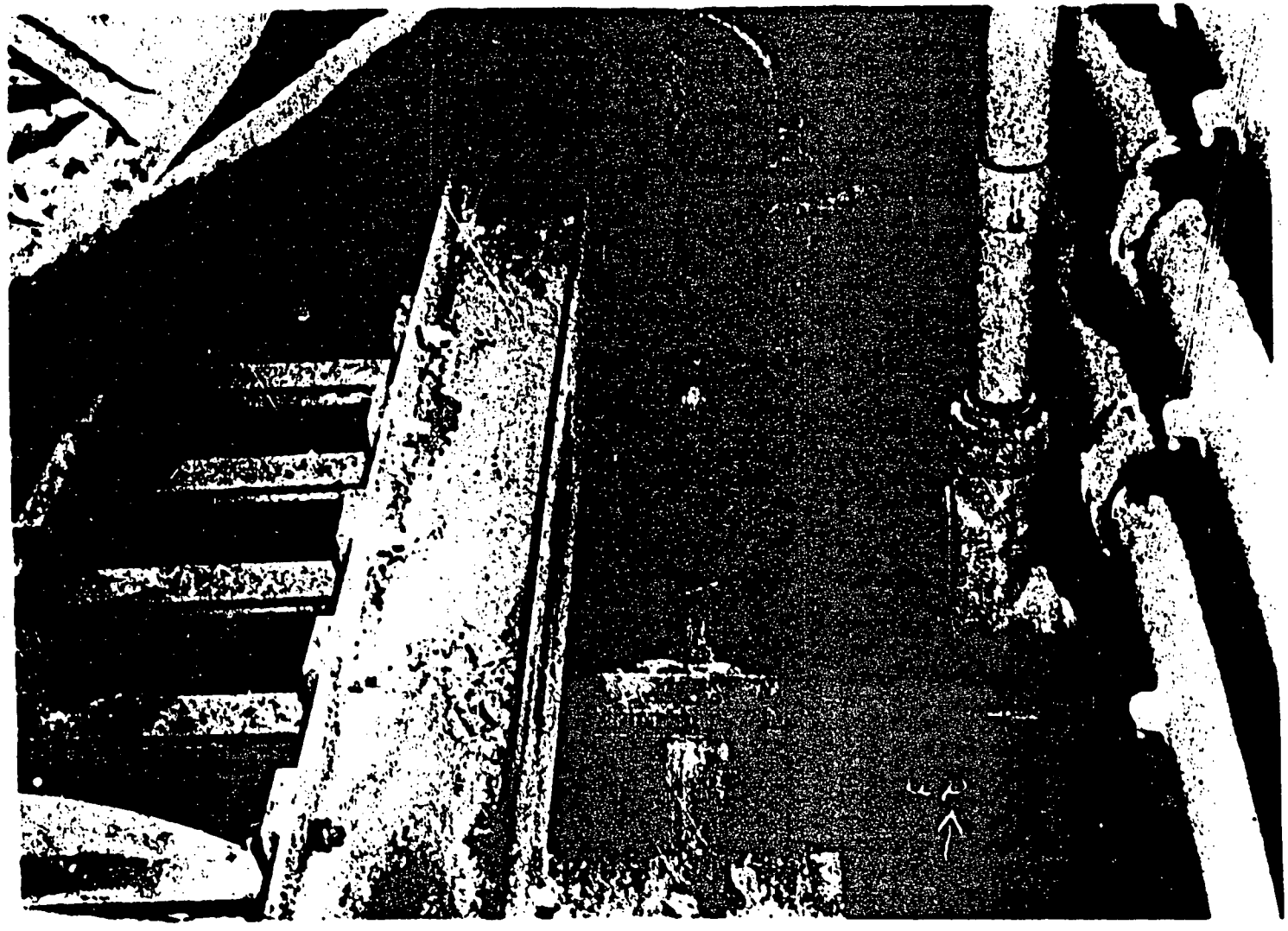
AREA OF FIRE



WH-K-86577-B


ASSUMED NORTH
PHOTOGRAPH 89438K
VIEW OF FIRE DAMAGE TO CONDUIT IN NORTHEAST CORNER OF REACTOR BUILDING
CONTROL ROOM EL 6120
REACTOR BUILDING EL 6210
4" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
4" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL (TYPICAL)
3" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 3930
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE



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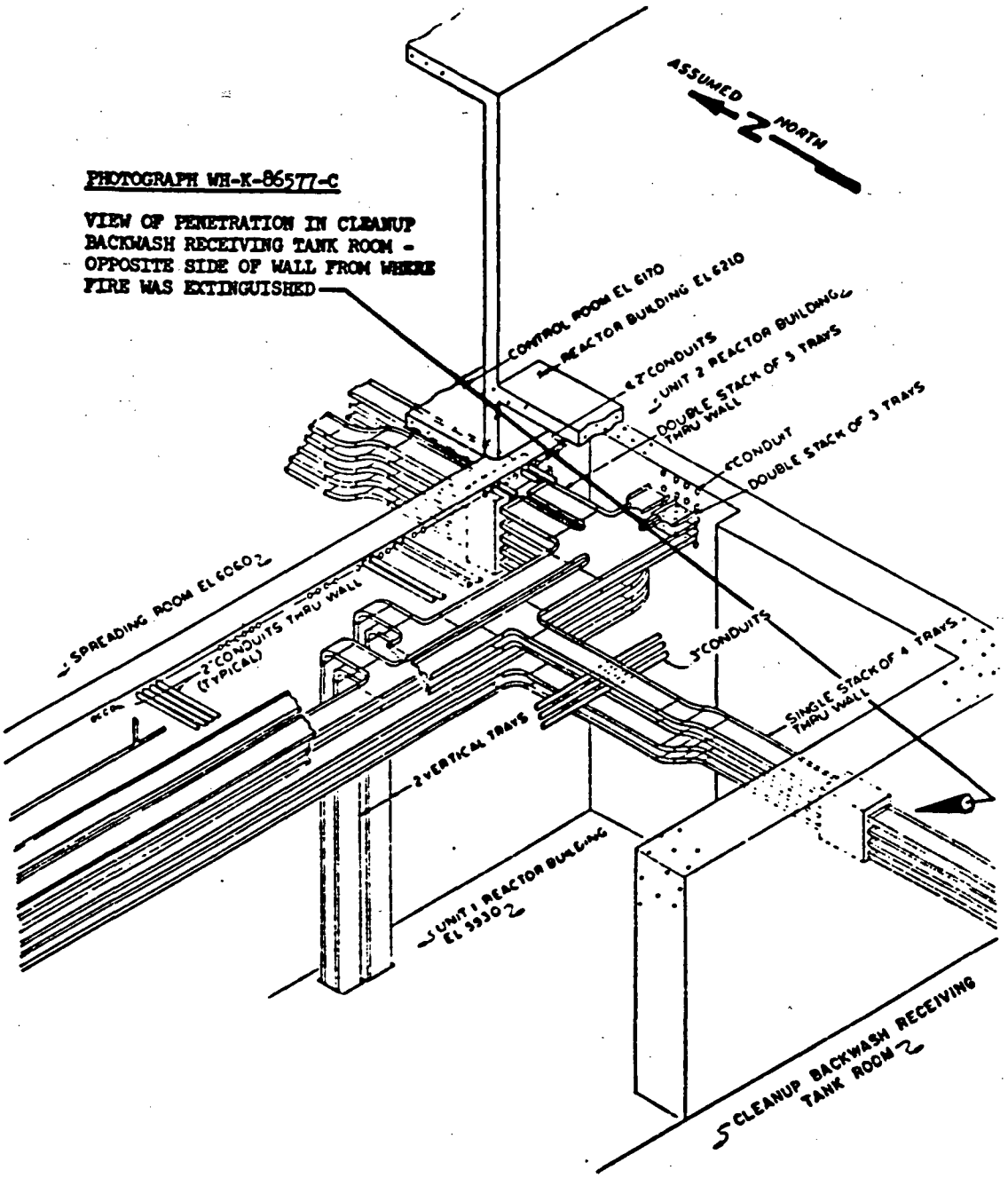
ASSUMED NORTH
PHOTOGRAPH WH-K-86577-C
VIEW OF PENETRATION IN CLEANUP BACKWASH RECEIVING TANK ROOM - OPPOSITE SIDE OF WALL FROM WHERE FIRE WAS EXTINGUISHED
CONTROL ROOM EL 6170
REACTOR BUILDING EL 6210
4" CONDUITS
UNIT 2 REACTOR BUILDING
DOUBLE STACK OF 3 TRAYS THRU WALL
4" CONDUIT
DOUBLE STACK OF 3 TRAYS
SPREADING ROOM EL 6060
2" CONDUITS THRU WALL
2" CONDUITS (TYPICAL)
3" CONDUITS
SINGLE STACK OF 4 TRAYS THRU WALL
2 VERTICAL TRAYS
UNIT 1 REACTOR BUILDING EL 5950
CLEANUP BACKWASH RECEIVING TANK ROOM

AREA OF FIRE

MH-X-965[illegible]-C

APPENDIX 8

INVESTIGATION REPORT BY THE NUCLEAR ENERGY LIABILITY AND PROPERTY INSURANCE ASSOCIATION (NELPIA) DATED MAY 1975

TVA's Browns Ferry Nuclear Plant
Athens, Alabama
NEL-PIA Policy (NF - 198)

General

During April 7-9, 1975, an investigation into the March 22, 1975 cable fire that occurred at the Browns Ferry Nuclear Station was conducted by J. A. Elsner and R. J. Kasper. During this three day period, an analysis was made of the events that occurred up to, during and after this untimely incident. As a result of this study, a report has been prepared summarizing what happened and why it happened. Applicable recommendations suggesting possible fire protection improvements have been developed for consideration.

Introduction

TVA's Browns Ferry BWR nuclear power plant is located on the Tennessee River south of Athens, Alabama. Completed in 1974, its Unit 1 became operational on August 1, 1974. Seven months later (March 1, 1975) its Unit 2 also became commercially operational. Since each Unit was engineered to produce 1100 megawatts (MW) of electricity, a total of 2200MW were being supplied to TVA's power grid when a fire started in the plant's cable spreading room.

Origin of Fire

The fire that started at 12:35 P.M. on March 22 began in the Cable Spreading Room (CSR) for Units One and Two. This room, at elevation 606 ft., is located directly under the Control Room (CR) for Units One and Two. Constructed of reinforced concrete the CSR is cut-off from (1) the CR by a floor having a fire resistance rating of at least 3-hours and (2) the Reactor Building by a 2 foot, 2-inch thick reinforced concrete wall of at least 8-hours fire resistance rating.

Story of Loss

The night before the loss occurred, workmen were pulling cable from the RB into the CSR through a 2" x 4" hole in the 26-inch thick concrete wall separating the Reactor Building from the Control Building. To understand the procedures followed imagine a 4-ft x 4- ft wall opening containing ten 18-inch wide x 4" high galvanized steel cable trays. Tiered five high, the parallel runs of tray are sealed on the RB side of the wall to prevent flame propagation from one fire area to another. Whenever new cable has to be pulled, an opening is made through the fire stop construction. Upon completion, the opening is resealed first using a pillow-type foamed polyurethane for backing. To both sides of this backing material is next applied a 2-3 inch thick layer of foamed-in-place polyurethane or silicone rubber (currently used) followed by a 1/4-inch layer of Flamemastic 71A to complete the seal.

At the time of the loss workmen were in the CSR checking for air leaks into the RB. Apparently, the Standby Gas Treatment System's capacity (one complete air change/day) was being overtaxed from excessive air leakage into the RB. Using a 10-inch candle the cable tray penetrations were being checked. By holding a candle close to the penetrations, the workmen observed if the candle flame would flicker or not. The CSR is maintained at a positive pressure with respect to the RB to insure that air flow is towards the RB whenever radiation escapes.

When examining BE cable tray penetration, the candle flame was drawn into a hole not sealed airtight by the pillow-type foam material and not yet coated with foamed polyurethane and Flamemastic. It immediately ignited the pillow-type cushioning material and chips of rigid polyurethane foam which in turn ignited plastic insulated control cables. Without portable fire extinguishers immediately available nor a "fire watch" on the other side of the fire wall, the fire began to burn uncontrolled.

Not able to put out the fire by hand using a flashlight and a bundle of rags, the workmen hurried to find some extinguishers. Having located one dry chemical and three CO_2 extinguishers, they attempted to extinguish the fire but found that the burning cable would repeatedly reflash once the extinguishers were emptied.

When hunting-down extinguishers to fight this fire, the public security officer was told of a fire in the CSR. He notified the shift engineer in the Control Room who called in the station alarm by dialing 299 on the telephone. Firefighting personnel responded to this alarm and proceded to extinguish the fire in the CSR and the RB.

The difficulties encountered putting out the fire in the CSR prompted the brigade to activate the Cardox total flooding system. After having experienced numerous problems with this system, the CO_2 system was finally discharged sometime after 1:20 P.M. Dumped three times during the fire, it controlled the fire at its source in the CSR and prevented a catastrophe from occurring.

The fire also spread rapidly into the Reactor Building through the opening in the cable tray seal. Hampered by smoke and the location of the fire, firefighters wearing breathing apparatus used several dry chemical and carbon dioxide fire extinguishers. Discharging them from scaffolding and ladders, they encountered continual reflash. The inaccessibility of the multi-tiered cable trays located 20-25 feet overhead, the occasional dripping of flaming insulation and the failure of the building's ventilation system compounded the firefighting effort.

At two o'clock firefighting efforts in the Reactor Building were abandoned in favor of an orderly station shutdown. For the next <u>two hours</u>, the cable fire was allowed to burn uncontrolled for lack of manpower, lighting and visibility. Once Units One and Two were shutdown, the firefighters resumed fighting the stubborn fire with portable 15-and 20-lb extinguishers.

Up until 6:00 P.M. multiple fires continued to burn in the cable trays. Continually, extinguishers were discharged onto the fires, often the fires would go out and repeatedly the fires would reflash. This phenomenon required a total of 51 fire extinguishers (20-15 lb. CO_2, 31-20 lb. dry powder "purple K"), 2-125 lb. "purple K" carts and 1-100 lb. CO_2 cart!

At 6:00 P.M. a decision was made to use water on the fire. The ineffectiveness of the portable extinguishers and the dust concentrations created by the discharged dry powders finally prompted firefighters to use a 1 1/2" hose stream. Backed by 3-2500 gpm @150 psi electric fire pumps, its use by the Athens Fire Department was found to be extremely effective, and within fifteen minutes or so the fire was essentially out.

<u>Damage</u>

Aside from corrosion damage and clean-up, the cable tray systems received the most damage. Approximately thirty 18-inch wide cable trays or sections thereof were damaged by fire. Assuming there were 100 cables in each of the trays involved in the fire, it is estimated that 60,000-65,000 feet of control and thermocouple cable were destroyed!

The loss of this amount of cable affected or impaired the operation of numerous systems. For example, the manner in which the cable failed simulated a loss of coolant accident. Steam isolation valves closed, motor operated valve control was lost and 250 volt D.C. power controlling the operation of circuit breakers failed. Also affected were numerous control boards, core spray pumping systems, exhaust systems, M-G sets, 4KV bus, and lighting circuits.

The repair, replacement and rearrangement of the cable trays and cable sections will require approximately three months. If Units One and Two are shut down this long, the TVA has estimated it will cost an additional $44M in fuel costs to supply sufficient power to its grid.

Cable Spreading Room, Similar Rooms and Other Areas

The Cable Spreading Room at Browns Ferry was considered to be poorly designed from an operational and fire protection standpoint. First of all it was engineered to house the cable spreading rooms of Units One and Two. Normally one would expect each unit to have its own cable room constructed in such a manner that a fire in one would not spread to the other. Were such an incident to occur that shutdown both units for months at a time, the repercussions would be far-reaching.

The congestion in this CSR was inexcusable. Cable trays were situated in every direction conceivable. The close proximity of horizontal and vertical arrays from floor to ceiling the length of the room (150 ft.) is not possible to relate in terms of the severity of the problem.

Such a massive array of cable trays in the absence of aisles voids any realistic fire-fighting effort without subjecting firemen to possible injury or even loss of life. To go into the heart of this CSR full of smoke and acrid fumes to put out a fire even wearing self-contained breathing apparatus would be considered beyond the realm of heroism. Most firefighters would ultimately resort to a manual defense of the fire outside the room using hose streams and high expansion foam.

Unless automatic fixed fire extinguishment is provided to protect the hazards inherent in these rooms, a loss beyond imagination should be anticipated. To overcome such a catastrophe, a first class, designed and approved, automatic fire protection system is needed.

Of the systems currently available, it is recommended that automatic sprinkler protection be provided. It offers the most effective and efficient way of putting out a burning cable fire. By wetting-down and cooling the red hot conductors, the Class A fire can be readily extinguished without much concern over reflash.

Although total flooding high expansion foam, halon and carbon dioxide are considered acceptable extinguishing agents by some agencies, they do not possess the coolant properties so necessary to ultimately put out the fire. Without their presence, continued reflash of the cables can be expected until the temperature of the conductors is well below the auto-ignition temperature of the cable insulation and jacket

Recommendations

Design

1. A CSR for each unit should be provided. Each spreading room should be cutoff and arranged totally independent of other CSR's by a fire barrier wall of 3-hours fire resistance rating.

2. At least one 3-foot wide 8-foot high aisle should be provided the length and width of the CSR to insure firefighting access.

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Class A fire doors should be installed at each entrance (at least two) into the room.

Fire Protection

1. A standard installation of open head, water spray sprinklers controlled by an automatic deluge valve and products-of-combustion acutated detectors should be provided in each CSR. The deluge valve should be located outside the room and connected to the station's annunciator system.

2. One-inch hose connections, equipped with 75 ft. of 1 1/2" woven-jacket lined fire hose and adjustable spray nozzle should be provided in the CSR and located at approximately 100 ft. intervals.

3. The concrete floor should be pitched to drain the sprinkler and hose stream discharge to a suitable drainage facility.

4. Approved smoke and heat venting of the CSR utilizing a powered mechanical exhaust system actuated by products-of-combustion detectors should be provided.

   Note: These recommendations are also pertinent to Cable Penetration Rooms and other areas where concentrations of cable exist.

Cable Constructions

The widespread use of combustible nonmetallic cable constructions in nuclear power stations has introduced a fire hazard that is not sufficently appreciated by station operators. Unless those in authority begin to recognize the seriousness of this hazard and its consequences, losses involving millions of dollars can be expected.

The premise that grouped nonmetallic cables will not burn or propagate flame has been a fallacy often promoted by certain interest groups. Up until recently users of electrical cable often believed that their tray cable was "nonburning", "nonpropagating "fire resistant" or "self-extinguishing". Not so! Recent cable fires and cable tray tests have substantiated that grouped cables do burn and are in fact a recognized fire hazard.

Often times manufacturers, designers and users will specify the pass-fail criteria of a cable construction to justify its intended use. When they do, flame resistance properties are often one of the first sacrificed for improved moisture, corrosion, oil, radiation and ozone resistance. The result is a cable construction that will burn in almost every conceivable tray configuration.

The corrosivity and toxicity of burning cable are also often ignored when selecting a cable construction. Whenever polyvinylchloride, neoprene or hypalon cables are burned, hydrogen chloride (HCL) gas is evolved. Combined with the water vapor in air, the hydrochloric acid produced will contaminate building steel, electrical equipment and instrumentation. The result is a major corrosion loss.

At Browns Ferry an assortment of cable constructions were noted. In Table No. 1 most of the types of cables involved in the fire are listed. All of the cables are 600 v. control cable or thermocouple cable. Typically, the cables are two conductor (2C) No. 16 and occasionally one conductor (1C) 300MCM and three conductor (3C) No. 16.

All of the cables consisted of a polyethylene (PE) or cross-linked polyethylene (XLPE) insulation and a polyvinylchloride (PVC), hypalon or nylon overall jacket.

Considered combustible by manufacturers and some users, these constructions have too often misled people that they do not burn. In fact, they do burn. Certain fire test and numerous fire losses have verified this.

To be more specific, the TVA decided that cable specimens passing Underwriters Laboratories Standard No.83 are acceptable for use in their stations. In the vertical test, a Tirrill gas burner is applied to a 12-inch specimen for 15 seconds then removed for a minimum of 15 seconds. This procedure is repeated five times. Cables that flame or glow longer than 60 seconds after any of the flame applications are judged not acceptable.

Reviewing the flame-retardant requirements of cable at Browns Ferry, all single and multi-conductor cables No.8 and larger were required to pass the misleading vertical flame test. All cable No.9 and smaller were required to pass the even more misleading IPCEA horizontal test. Suprisingly, single and multi-conductor cable having a nylon jacket were not required to pass <u>any</u> flame test!

Because these small-scale tests can be very misleading, especially for grouped multi-conductor control and power cable, users should insist that stringent cable fire tests indicative of "real-life" situations be required. One test currently being used by some utilities is the IEEE No.383 flame test; others, such as NEL-PIA/MAERP, insist that fully loaded, tiered trays be tested, using more severe ignition sources.

In summary, minimal consideration was given to the obvious burning characteristics of the cables used at Browns Ferry. Were there concern, it would have been evidenced by more stringent flame tests, better grade, truly fire retardant cable constructions and fixed fire protection for the cable tray systems.

<u>Recommendations</u>

1. A re-evaluation of current cable testing requirements should be made to establish the pass-fail criteria for flame propagation of "real-life" cable tray systems.

2. As a minimum requirement today only those cable constructions that will pass the current IEEE No.383 flame test should be used. Note: This does not infer that cables passing this test do not require fire protection nor that certification of cables passing more realistic tests are not essential for tommorrow's cables.

3. Whenever practical, cables that do not liberate copious quantities of corrosive gases should be used particularly in strategic, relatively inaccessible and highly susceptible areas.

Table No. 1

Cable Constructions

Manufacturer	Size	Cable Description
1. Boston Insulated Wire	2/C No. 16 7 strand tinned copper	30 mils XLPE insulation, .001" mylar tape, 46 mils overall hypalon jacket
2. Cerro	3/C No. 16 1969 WVR	20 mils PE insulation, 10 mils PVC conductor jacket or 26 mil PE insulation, 4 mil nylon conductor jacket, 80 mil overall PVC jacket
3. Continental Wire	2/C No. 16 1968 WUB	25 mils 90°C XLPE insulation, 30 mils overall hypalon jacket
4. General Cable	1967 600 v. CPJ	XLPE insulation, PVC overall jacket
5. Okonite	2/C, 3/C No. 16 7 strand	20 mils 90°C XLPE, 40 mils overall hypalon jacket
6. Plastic Wire & Cable	1971 600 v. PJJ 2/C No. 12	20 mils PE, 10 mils conductor jacket, 60 mils overall jacket
7. Simplex	600 v. 1/C 300MCM PNJ	PE insulation, 4 mils nylon jacket

Symbols:

XLPE - cross-linked polyethylene

PE - polyethylene

PVC - polyvinylchloride

PJJ - PE insulated, PVC jacketed, single conductor

CPJJ - XLPE " , PVC " , multi-conductor

PNJ - PE " , nylon " , single conductor

CPSJ - XLPE " , shielded, PVC jacketed, single conductor

Cable Tray Protection

The cable trays involved in the fire were not protected by automatic or manual fixed fire protection systems. The only fire protection equipment immediately available were a 1 1/2-inch hose connection and a 20 lb. dry chemical extinguisher which was ironically never used.

Having already emphasized the combustibility of the cables involved in the fire, the configuration and inaccessibility aspects of the cable trays should also be studied. Consider 18-inch wide, open ladder trays with rungs every six inches stacked vertically five high with only six inches of vertical separation between trays. Next, imagine another vertical array of trays similiar to the first six inches away. Finally visualize this arrangement 20-25 feet above the floor in a windowless area with access from only one direction.

Such a large and important arrangement of trays filled with plastic jacketed cable is one that should have been protected by a system of automatic sprinklers. The presence of sprinklers would have effected control and extinguishment of the fire within minutes. Once discharged, the water would have sufficiently cooled the copper conductors preventing cable reflash. The amount of smoke generated and corrosive gas liberated would in turn have been considerably less.

Recommendations

1. Cable tray systems should be protected by automatic, zoned, open-head, water spray sprinkler systems arranged to discharge directly onto the cables in the trays.

2. An approved fast acting products-of-combustion type detection system should be provided to actuate the deluge system having sectional control.

3. Adequate floor drains and curbs should be provided to safely remove discharged sprinkler water. Drainage water should be monitored for radioactive materials before being released to the environment. Curbs should be provided around all floor penetrations.

4. Approved noncombustible fire stop constructions should be located in each cable tray and spaced at maximum intervals of 10 feet horizontally and 10 feet vertically. Note: Cable derating should be given consideration when installing fire stops.

5. Wherever practical, isolate, shield, or relocate water damageable equipment.

Consequential Corrosion

Whenever PVC, neoprene and hypalon cable constructions are involved in a fire, frequently the fire damage is insignificant compared to the consequential damage caused by corrosion. The hydrogen chloride gas liberated by these burning materials readily combines with water from the combustion process and water vapor in the air to form hydrochloric acid. When this acid comes into contact with a metal surface, it will react with it to form conventional rust. When it comes into contact with building materials having a calcium element, the hydrochloric acid will react with the calcium constituent causing flaking, tensile strength reduction and steel reinforcement corrosion.

The introduction of such acids into a nuclear station is certainly not advisable. The damage caused by HCL followed by progressive corrosion is often so devastating that months and millions of dollars are required to reactivate a unit shutdown by a fire of relative unimportance.

At Browns Ferry, there was considerable evidence of corrosion damage in the Reactor Building. Throughout elevation 593, stainless steel piping, copper tubing, instrumentation, electrical relays, conduit, valves, pull chains, sleeves, bolts and nuts were found rusted or coated with a brown film.

At elevation 620', there was also evidence of corrosion damage to equipment. Had the open steel-grate stairwells and numerous floor openings been enclosed or sealed air tight, it is doubtful corrosion damage and carbon deposits would have spread to this floor elevation.

Although the TVA does not know how much the cost will be to repair or replace all the equipment damaged by corrosion, it is conceivable that it will take at least three months and cost millions of dollars. Had sufficient consideration been given to this possibility when cables were initially selected, it is doubtful that corrosion would have been a major factor in this loss.

Measures that should be taken to reduce the risk of corrosion damage have been outlined in other sections of this report. Basically, plastic cables having corrosive by-products should be avoided, floor penetrations should be tightly sealed, automatic fire extinguishing systems should be installed, and ventilation systems should be arranged to localize the fire and minimize the concentration of corrosive gases.

Appraisal of Firefighting Concept

Three sections in the "International Guidelines for the Fire Protection of Nuclear Power Plants" should be referenced at this juncture. They are: 1) Appraisal of firefighting concept 2) Mobile firefighting and 3) Equipment of the plant fire brigade. These sections not only emphasize the importance of automatic fixed fire protection installations, they equally stress the importance of a coordinated effort between the local fire department and station fire brigade.

Unless a previous agreement has been reached amongst station personnel and public fire authorities, a situation is apt to present itself that will result in a misguided and costly firefighting approach. For every conceivable loss, a detailed action plan and firefighting approach should be developed in close cooperation with the insurance company and other competent authorities.

Although there is some merit to the firefighting procedures used at Browns Ferry, another approach using water as the extinguishment is often considered more effective. Since plastic electrical cables are Class A materials when they burn, extinguishing agents compatible with Class A fires should be used, e.g. water or light water. Extinguishing agents including carbon dioxide, halon, B:C dry powders and foams that do not have a cooling effect are normally ineffective and should not be used.

For a typical Class C electrical fire, with voltages of 600 volts or less to ground, there is virtually no danger to firefighters directing hose streams on cables. This is especially true of cable tray systems located 20 feet or more above the floor.

Were this concept employed soon after the fire started, it is postulated that the fire would have been extinguished in a matter of minutes. Cable damage would have been minimal, corrosion damage would have been insignificant and an orderly station shutdown would have been possible. In addition, the Reactor Building's ventilation system would not have been impaired and conceivably redundant circuits would not have failed.

Indoor Hose Connections

Although a 1 1/2" hose connection equipped with 100 ft. of woven-jacket fire hose was located 66 feet from the origin of the fire, its use was not as effective as it could have been. First of all, the decision not to use water during the first five hours of the fire proved costly. Were water used from the fire's inception there is little doubt that the fire would have been out without reflash within minutes and that cable and equipment damage would have been minimal.

When the decision was finally made to use water on the fire, plant personnel were reasonably assured that all electrical power was off in the fire area. The Athens Fire Department proceeded to attack a small vertical cable tray fire to see the water's effect. Noting that it worked fine without electrical arcing, they proceeded to attack the fire in the stacked upper trays located 20-25 feet above the floor.

Adjusting the spray nozzle they observed that the water spray wouldn't reach up into the trays. Firemen decided to use a combination spray-straight stream nozzle on the fire truck. Removing the adjustable spray nozzle, they quickly learned that the hose line threads were not compatible with those of the fire department's nozzle. The nozzle flew off!

Ultimately the firemen used ladders and scaffolds to get to the seat of the fire with the hose line. Firemen reported only 2-3 feet visibility, blue flames in the trays and no electrical arcing. Within minutes the fire was out although much localized steaming was noted.

Recommendations

1. Fire protection equipment including hose, nozzles, standpipe valves, and hydrants should have compatible threads with existing equipment and the local fire department.

2. Combination spray-straight stream nozzles should be provided on each hose connection to effectively combat Class A fires normally inaccessible e.g. cable tray fires.

3. Standpipe risers should be sealed on each floor to prevent smoke and corrosive gases from penetrating into areas normally unexposed to the effects of fire.

Smoke and Heat Removal

The effective removal of toxic gases and heavy concentrations of smoke is essential when combating a fire in a nuclear station. The primary reason for heat and smoke venting is that it enables firefighting personnel to gain early access into the fire area to extinguish the fire, to shutdown the station in an orderly manner and to perform emergency rescue work. In addition it will help to reduce smoke and acid vapor contamination.

At Browns Ferry, the smoke and acrid fumes generated by the burning cable were so intense that firefighters were required to wear self-contained breathing apparatus. Without such equipment on hand there was no possible way to manually combat the fire. Fortunately, plant personnel properly saw the need to acquire a supply of breathing apparatus just in case such a situation arose.

Had the ventilation system for the Reactor Building continued to operate unimpaired, the fire would not have burned for seven hours. Instead, within fifteen minutes of the start of the fire, the ventilation system shutdown. Apparently its power supply was located in one of the cable trays involved in the fire.

Running temporary wires to bypass the damaged cables, the ventilation system was turned back on at 4:40 P.M. Allowed to run for twenty minutes, it helped to clear the area of smoke and noxious gases and allow firemen to evaluate their next move. At 5:00 P.M. someone ordered the ventilation system shut down reportedly because it was helping to spread the fire.

Recommendations

1. Approved smoke and heat venting facilities independent of the station's normal ventilation system should be provided throughout areas having a combustible occupancy. Each system should be actuated by products-of-combustion detectors and arranged to contain the release of radioactive materials.

2. The mechanical exhaust system should be powered from electrical feeders run outside the fire area. If inside wiring is necessary, mineral-insulated metal sheathed cable should be used.

3. Additional preventative measures outlined in the "International Guidelines for the Fire Protection of Nuclear Power Plants" should be implemented. Specifically, sections 6.1 (extraction of smoke and heat) and 6.2 (preventing corrosion) are applicable.

Cable Penetrations (Trays, Conduit)

Whenever cable is installed in trays or conduit, wall and floor penetrations are inherently required. To insure the integrity of the walls and floors through which these systems pass, penetration seals are needed. Normally, approved seals that will maintain the fire resistance rating of the wall or floor are required by the authority having jurisdiction.

To Monday - morning quarterback which penetration seal specification should have been used in the Cable Spreading Room is not the intent of this report. What is important is an awareness that Underwriters' Laboratories and Factory Mutual are currently listing and approving devices and constructions for wall openings. Subjected to the ASTM E119 fire test, wall assemblies designed to protect electrical cables and/or conduits against the passage of flame and smoke are being tested.

Assemblies should be tested in conjunction with their intended use. If used in a wall having a 3-hour fire resistance rating, a wall opening protective having a rating of three hours should be installed. Likewise, if mounted in a floor opening, a device listed specifically for floor openings should be used. If a protective seal that is being considered for use in walls and floors is not currently listed or approved, the manufacturer should be requested to have his product tested according to the E119 test procedure by a recognized testing laboratory.

It is also important that wall and floor opening protectives be installed and tested without risk of fire or personnel injury. Installation and final test procedures

should not allow the use of open flames nor should they permit temporary unprotected openings overnight and over the weekend. Sufficient precautions should always be taken to insure that all wall and floor openings are sufficiently sealed to prevent the spread of fire from one fire area to another.

Recommendations

1. All wall and floor openings through which electrical cables or conduits penetrate should be protected against the passage of flame and smoke by devices and constructions approved or listed by recognized testing laboratories.

2. Temporary wall and floor openings should be sufficiently sealed with a noncombustible material at the end of each working day to insure the fire integrity of the wall or floor.

3. Open flames should never be used to check the installation, gas tightness and integrity of penetration seals. Whenever protected openings are examined, fire extinguishers should be immediately available to those checking for openings on both sides of the wall.

Self-contained Breathing Apparatus

Discussions with station personnel that fought the fire uncovered a wealth of information about the use and effectiveness of self-contained breathing apparatus. First of all, it was learned their use was imperative to attack the fire, manually shut down Unit One and continue working in the Control Room. To also emphasize how invaluable they were, the fire would have burned uncontrolled for several more hours had they not been used. An orderly station shutdown would have been jeopardized and personnel would have been subject to possible injury including death.

Certain problems concerning their use were noted. Their service life was not adequate for the time they were able to get to the seat of the fire. Continually firefighters had to leave the fire to replenish their air supply having been in the fire area only ten minutes. To shut some valves manually it often required men wearing Scot Air Paks 2-3 trips. Those with long hair often experienced smoke inside their mask from a poor fit.

Overall Chem-Ox respiratory protective equipment was considered a better unit than the Scot Air Paks. They lasted longer and fit better. On the other hand some of the Chem-Ox units were found inoperative whereas the others all worked.

Recommendations

1. Self-contained breathing apparatus approved by the United States Bureau of Mines and described in NFPA No.19B should be provided for all firefighting and control room personnel. Preferably, their service or operating life should be one hour.

2. An on-site reserve air supply should be available and arranged to expediently replenish the supply of air in each unit so that the designed service life is available.

Physical Independence of Redundant Circuits

The loss of some of Unit One's redundant systems illustrates the importance of physically independent redundant circuits. Permitting redundant circuits in trays that adjoin other trays loaded with highly combustible cable is not a sensible nor safe practice. Ultimately a fire in one tray will expose and eventually involve cables in the adjoining trays.

NRC's Regulatory Guide 1.75 "Physical Independence of Electric Systems" contains specific separation criteria for Class IE circuits. Although the specifications in this Guide are an improvement over previous guidelines, they do not require that all redundant circuits be separated by a fire barrier wall. Instead, they permit IE circuits to be installed in locations commesurate with the potential hazard. Such analysis and judgment considerations often dilute the specific criteria intended.

In order to insure a safe and expedient station shutdown, redundant circuits should be separated from each other by a minimum three hour fire barrier wall. By arranging the circuits in this manner, a catastrophe involving one system should not conceivably interfere with the continued operation of the other. Such a requirement should be incorporated into the station's design during the planning stages.

The use of mineral insulated metal sheathed cable or highly flame retardant cable should be considered for one of the two redundant circuits. These cables will insure reliance of the electrical continuity of the primary circuit and additional security that an orderly station shutdown will take place.

Recommendations

1. All redundant Class IE circuits and the equipment served by these circuits should be separated from the primary Class IE circuits by a minimum three hour fire wall. This will require that a redundant cable spreading room be constructed.

2. Mineral insulated metal sheathed cable or equivalent fire resistant cables should be used in one of the two Class IE electrical circuits.

Cardox Total Flooding System

A 17-ton capacity Cardox System was installed to provide protection for the Cable Spreading Room. Arranged to operate manually, the carbon dioxide system was designed to discharge gas into the room for approximately 2 1/2 minutes if actuated electrically. If operated pneumatically the system was arranged to discharge CO_2 continuously until either the valve was closed or the supply was depleted.

At the time of this loss the workmen in the CSR had turned off the electrical power supply to each of the two Push-button Stations. By doing so, the system can not be operated electrically until the toggle switch (located on Unit Two side of the CSR) is turned back on.

When the decision was made to discharge the Cardox system into the CSR, the Assistant Shift Engineer went to the Unit One side of the CSR to operate the Pushbutton Station. It did not work. Noting that the power supply had been turned off (red light out), he attempted to trip the system pneumatically from the Electro Manual Pilot Control

Cabinet. Because a steel plate used in shipment to protect the glass over this cabinet had not been removed when installed, he could not break the glass to open the manual valve!

Next he hurried to the Unit Two side of the CSR, turned back on the power supply at the Relay Cabinet and finally tripped the Pushbutton Station that discharged the total flooding system.

Although flooding the CSR with CO_2 prevented a major fire from occurring inside, it should be noted that the system's capacity was the key to the minimal damage incurred. For some unknown reason, the ventilation system for the CSR had not been interlocked to shutdown when the CO_2 system discharged. When the gas entered the room the exhaust system continued to operate dissipating the CO_2 concentration. This caused reflash! Were the exhaust system interlocked to shutdown it would not have been necessary to discharge the CO_2 system three times.

According to discussions with station personnel, the Kidde ionization detection system in the CSR did not operate during the incipient stages of the fire. However, about the time CO_2 began discharging into the CSR, the detection system did in fact actuate, not from the products-of-combustion but from the CO_2.

Recommendations

1. The ventilation system in the CSR should be arranged to shutdown whenever the Cardox system is discharged.

2. The Cardox system should be rearranged to operate automatically upon actuation of the ionization detection system. Note: A one minute delay should be incorporated into the system to allow workers ample time to leave.

3. A written procedure and permit system should be adopted that would require employees to obtain written permission to impair fire protection equipment.

4. An acceptance test of the fire protection system, including a complete discharge, should be conducted and witnessed by the installer and authority having jurisdiction.

5. An investigation into the compatability of the ionization detectors with the products-of-combustion generated by the burning cable should be made to insure that the detectors will in fact operate during the incipient stages of a fire.

Control Room

Each time it was necessary to discharge the Cardox total flooding system in the Cable Spreading Room, CO_2 gas and smoke from the fire would enter the Control Room through unplugged holes in the floor. Two 4" x 16" and two 6" x 24" unprotected floor openings were found under the "electrical desk" in the Control Room. To compound this problem the gas and products of combustion that escaped through these openings forced the operators in the Control Room to wear Scot Air-Paks when shutting down Unit Two.

Recommendations

1. All floor openings between the CSR and the CR should be sealed air tight with a material that will insure the fire resistance integrity of the floor. Only penetration seals listed by Underwriters' Laboratories or approved by the Factory Mutuals should be considered. Cellular concrete, and inorganic assemblies as described in the "International Guidelines" may also be considered.

2. Self-contained breathing apparatus approved by the United States Bureau of Mines should be located in the CR to insure an orderly station shutdown and to minimize breathing hazards to personnel. The supply should be sufficient for the number of operators and the time it takes to effect a safe shutdown.

Stairwells, Vertical Openings, Mechanical Penetrations

It was disturbing to note the presence of unprotected steel-grated stairways and floor openings around steam pipes, standpipes and mechanical penetrations in the Reactor Building. Although fire did not spread through these openings into other areas, smoke and hydrochloric acid fumes did. An examination of the building structure and equipment on the floors above the fire revealed heavy carbon deposits on walls and ceilings and substantial corrosion damage to stainless steel piping, electrical equipment, and instrumentation.

Were all vertical openings between floors sufficiently sealed to prevent the passage of smoke and corrosive vapors, corrosion damage and general clean-up would have been confined to floor elevation 593. Instead, clean-up and repair operations have to be expanded to include those areas needlessly exposed to carbon deposits and acid fumes

Recommendations

1. All stairwells, elevators, chutes and other vertical openings should be enclosed in approved masonry towers with air-tight, automatic closing Class B fire doors at each opening into the building.

2. All unprotected vertical openings between floors (hoistways, steam pipes, etc.) should be sealed air-tight.

UL-NEL-PIA Cable Testing Program

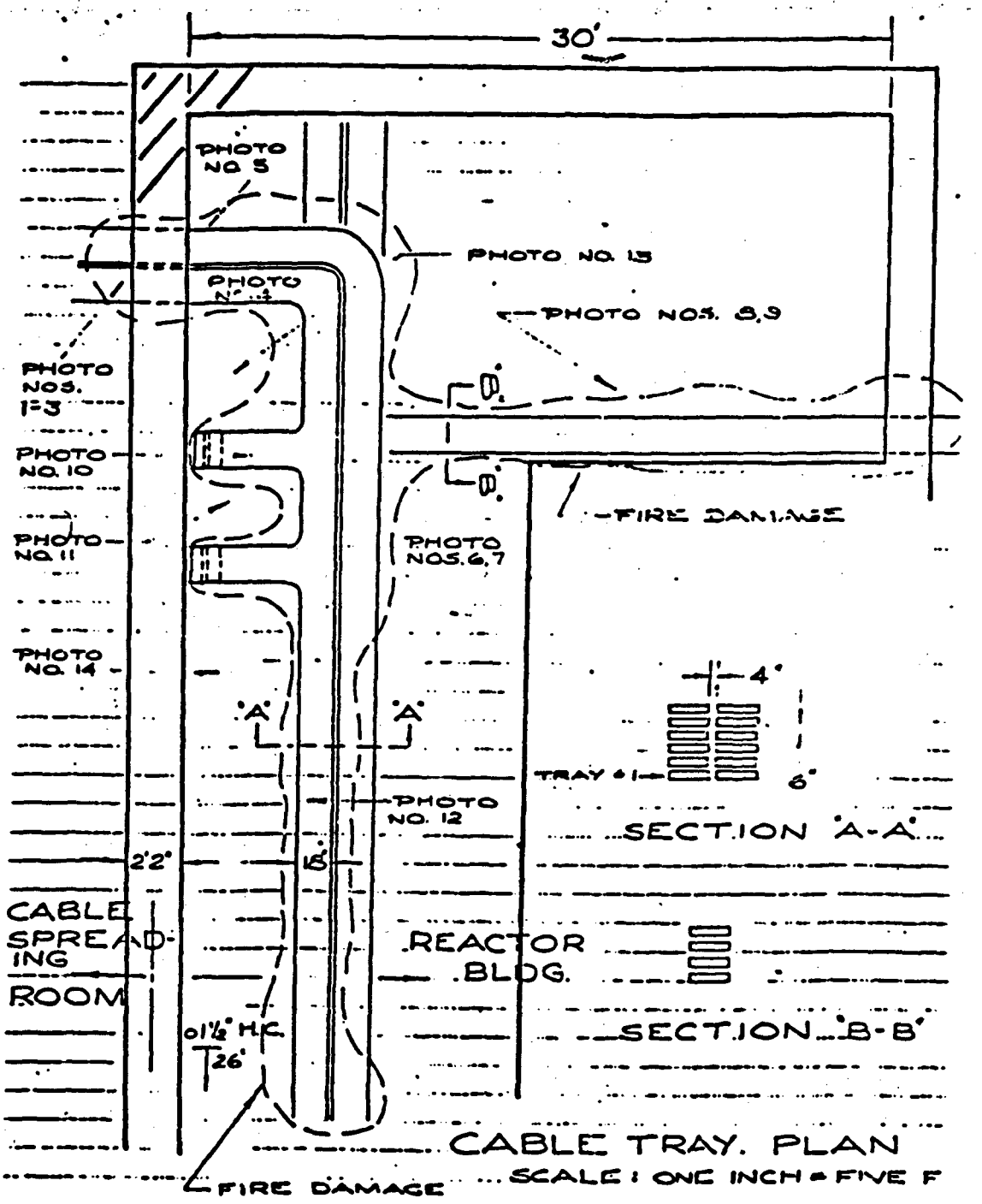
A NEL-PIA sponsored cable testing program is being conducted at Underwriters' Laboratories to determine the relative performance of cables when subjected to the IEEE Standard No.383 vertical flame test using a 70,000, 210,000 and 400,000 Btu/hour gas burner. Various control cable constructions will be tested vertically and horizontally in multi-tiered trays to determine the effects of these ignition sources on flame propagation and circuit integrity.

It is strongly recommended that the TVA consider including representative samples of cable involved in the Browns Ferry fire in this full scale testing program. Tested horizontally in multi-tiered trays and vertically, an understanding of the cable's "real-life" (not laboratory) burning characteristics can be demonstrated. In addition, the flame propagation and circuit integrity properties of cables considered highly flame retardant could be compared with those of the cables destroyed in Unit One.

Time Schedule of Events (Approximate)

Saturday, March 22, 1975

12:35 P.M.	Fire started in Cable Spreading Room.
12:40 P.M.	Fire Alarm 299 called in.
12:50 P.M.	Ventilation system in Reactor Building shutdown.
12:51 P.M.	Unit One reactor scrammed.
12:55 P.M.	Public Safety Service fire truck arrived.
1:02 P.M.	Unit Two reactor scrammed.
1:09 P.M.	Athens Fire Department notified.
1:20 - 1:30 P.M.	Cardox total flooding system discharged.
1:25 P.M.	Athens Fire Department arrived with one truck (chief plus six men)
1:30 P.M.	Reactor Building lighting lost.
1:30 - 2:00 P.M.	Self-contained breathing apparatus required in Control Room.
2:30 - 3:00 P.M.	Cardox system discarged the second time.
2:00 - 4:00 P.M.	Cable fire in Reactor Building burning unhampered. Firefighting effort abandoned in order to shutdown Units One and Two.
3:00 - 4:00 P.M.	Cardox system discharged for the third time.
4:40 P.M.	By running temporary wires, the ventilation system in the Reactor Building was reactivated.
5:00 P.M.	The Reactor Building's ventilation system was again shutdown.
6:00 P.M.	Hose stream first used on fire in Reactor Building.
6:45 P.M.	Fire considered out in Reactor Building.



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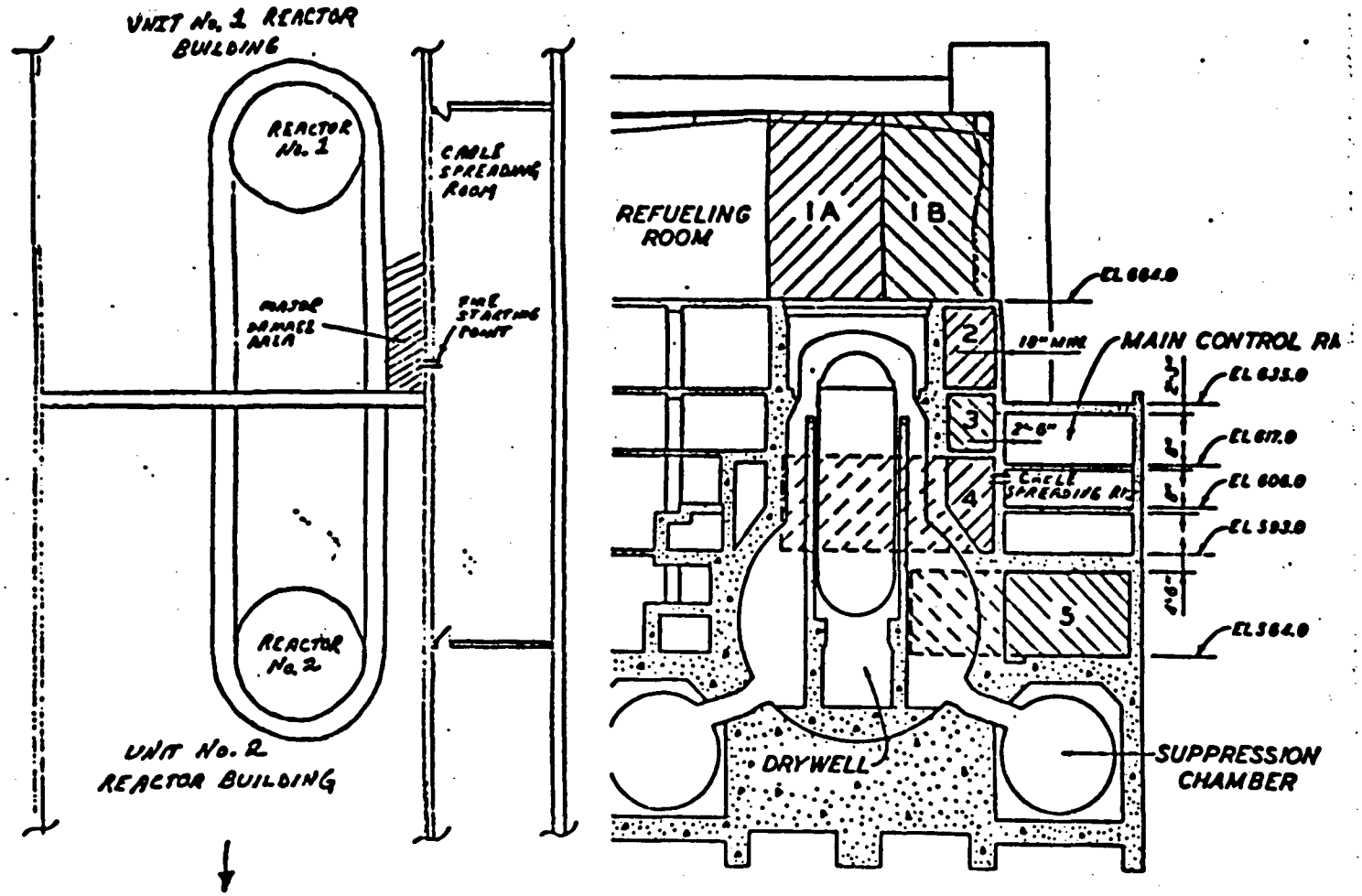
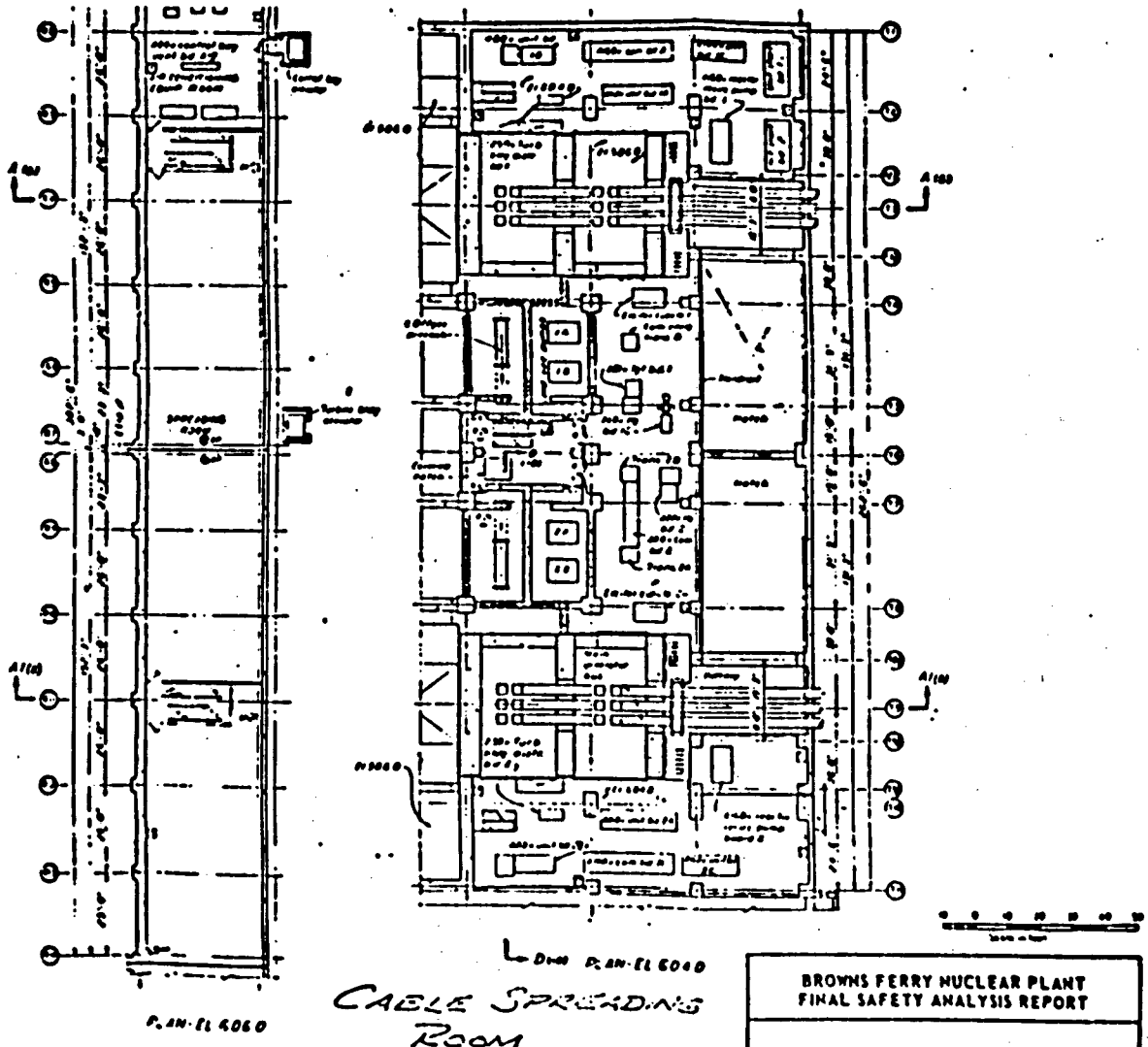


PLAN VIEW

SECTIONAL VIEW

FIGURE No. 1



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

Powerhouse Equipment Plans—
Elevations [illegible]

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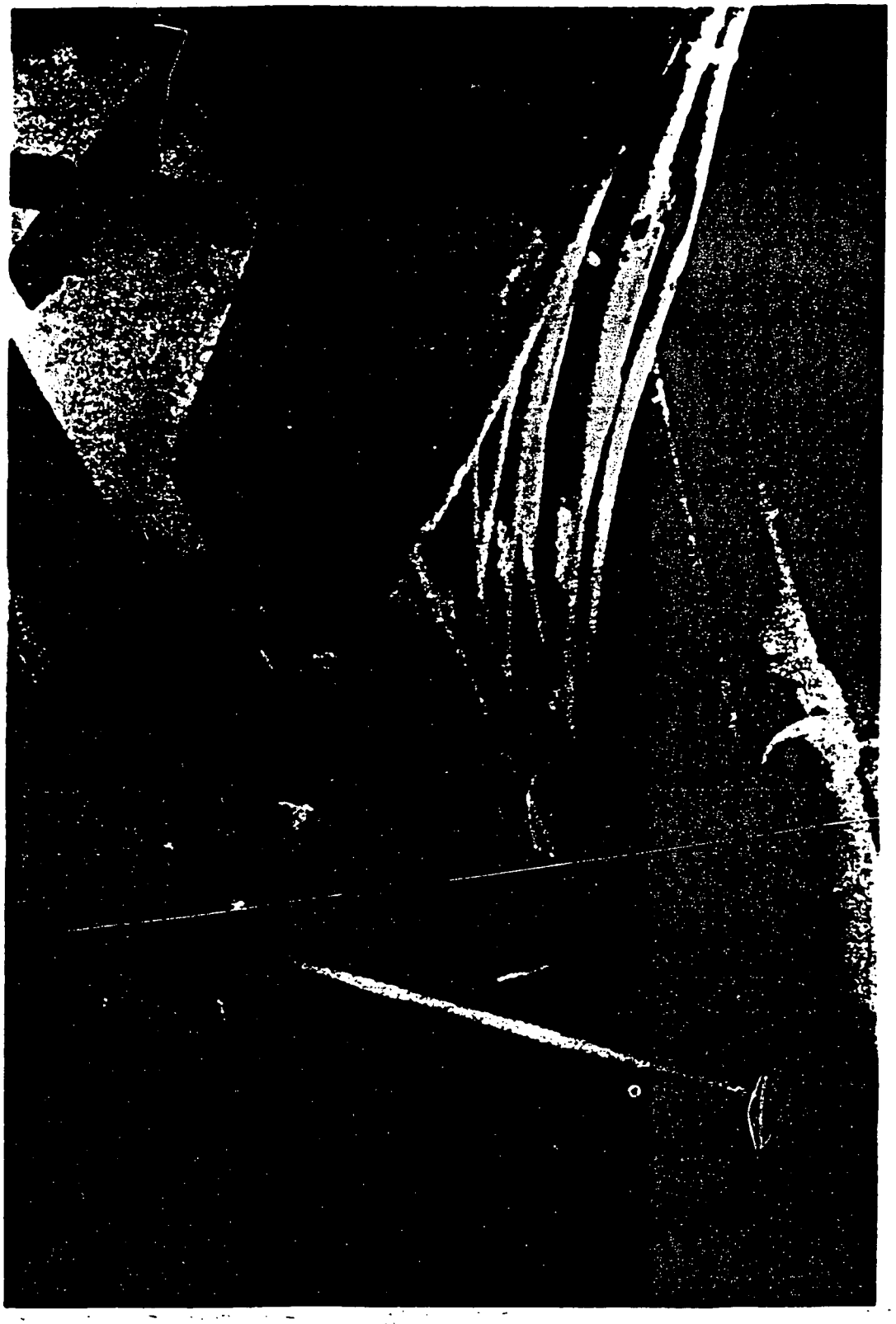
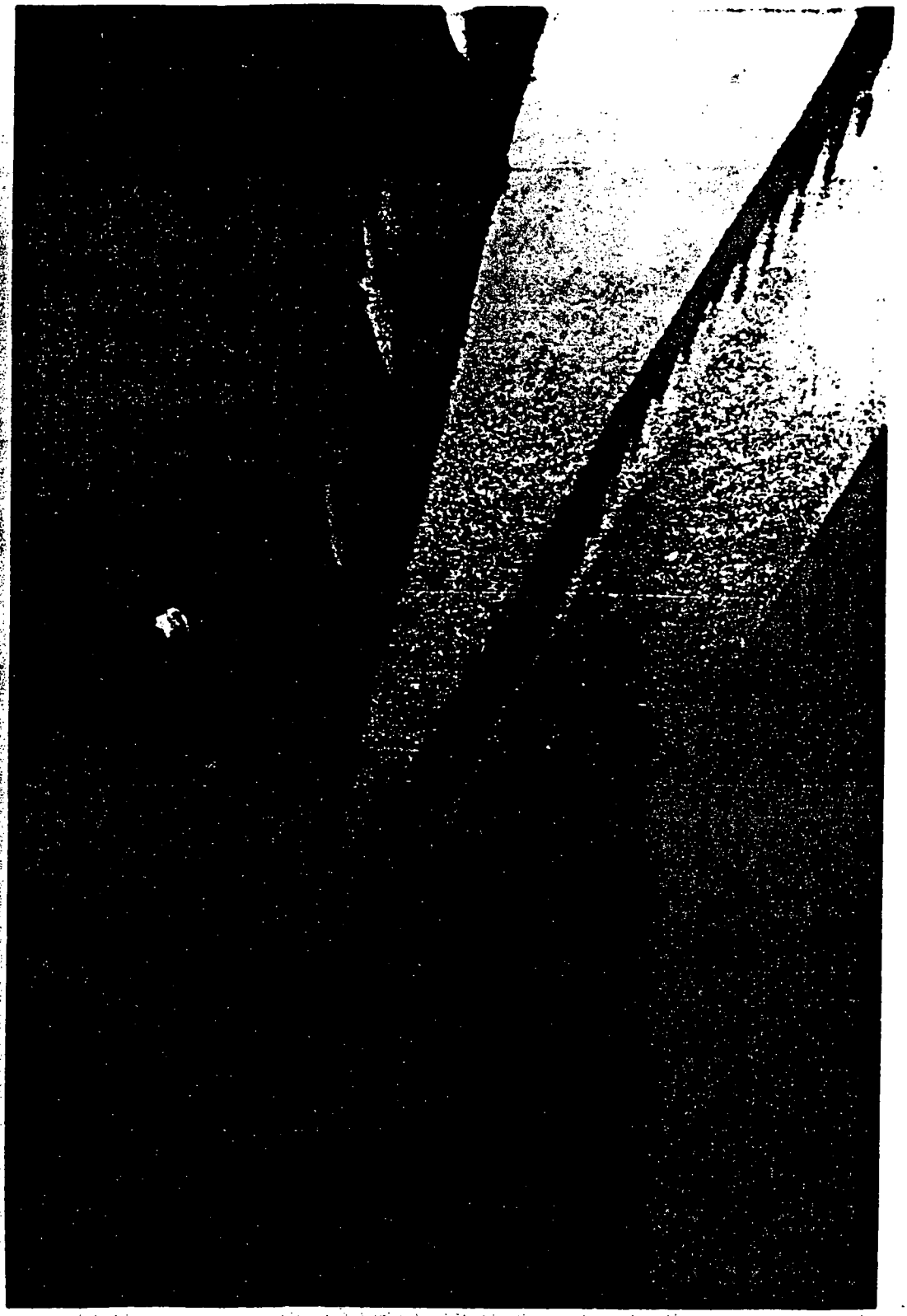
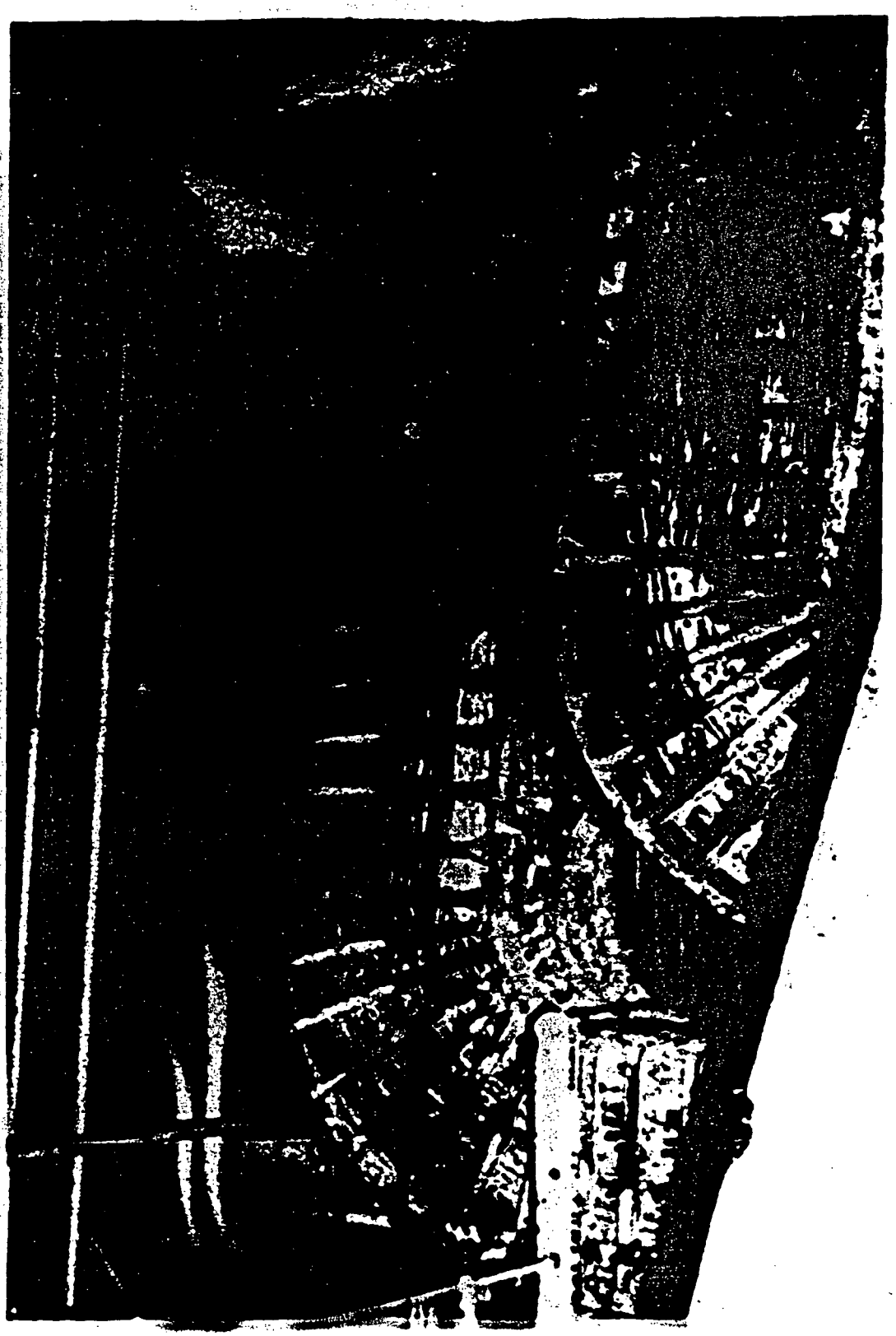
The fire originated in the middle tray. Taken inside the Cable Spreading Room, this photograph shows the cable trays passing through the wall into the Reactor Building.



Also taken inside the Cable Spreading Room, this photo shows the top three trays involved in the fire. Observe the exposed conductors in the top and bottom trays and the close proximity of the trays to each other. The white powder on the middle tray is from a dry powder extinguisher.



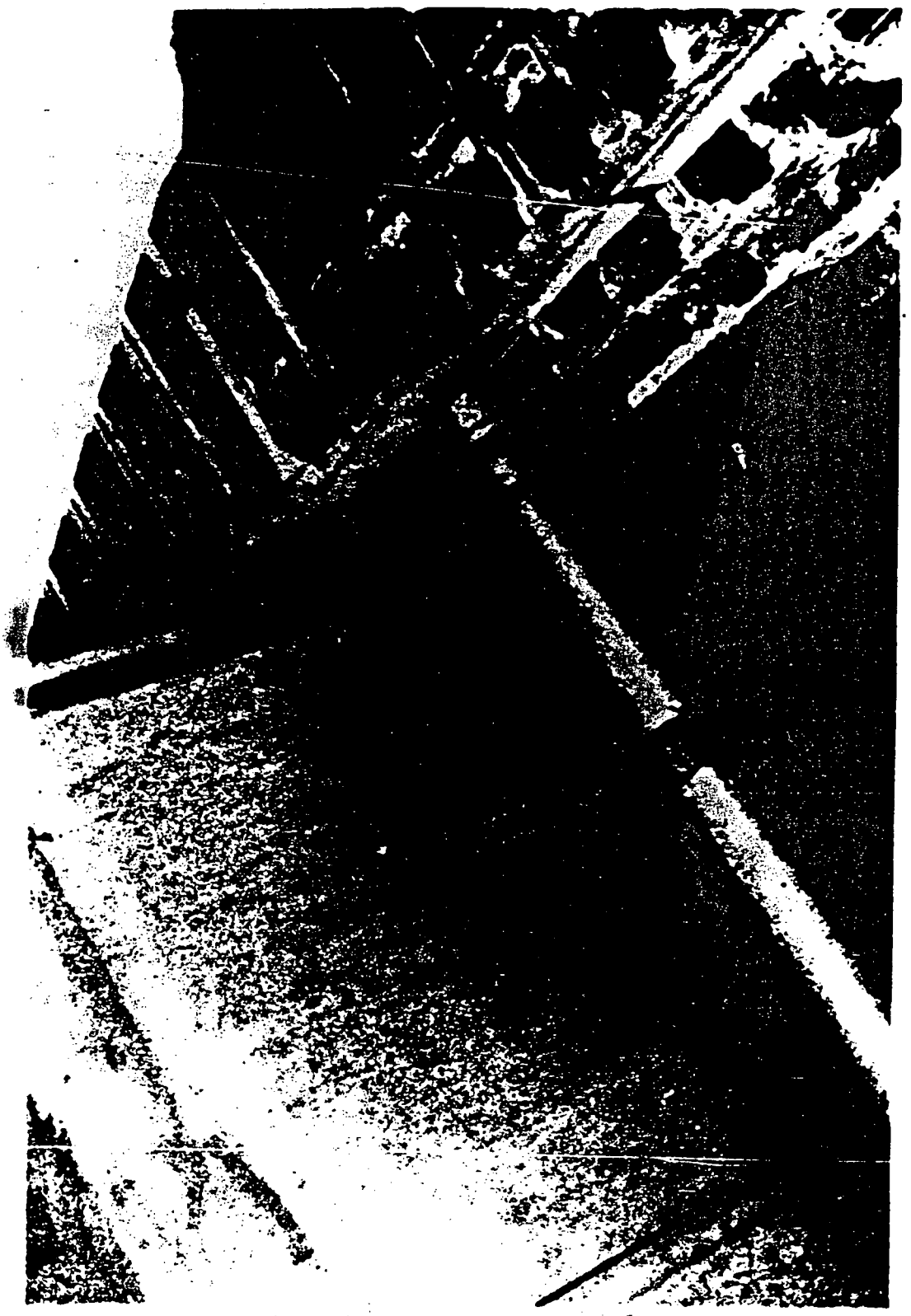
This is a close-up view of the top cable tray. This view clearly shows 1) the thickness of the wall between the Cable Spreading Room and the Reactor Building, 2) a portion of the fire stop and 3) the products of combustion from the burning cables in the top tray.



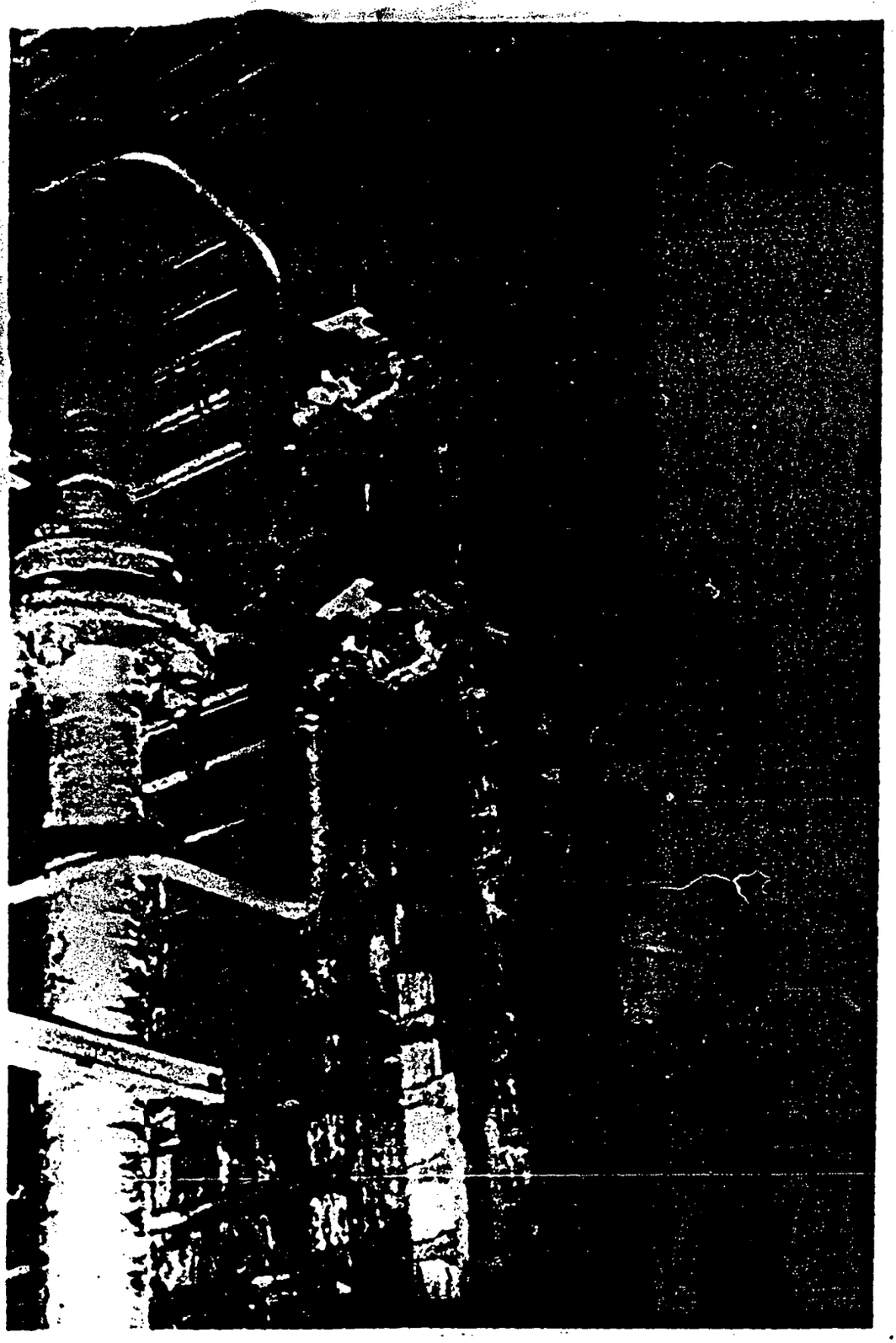
Looking up, this photo was taken in the Reactor Building. It shows the cable trays coming through the fire wall separating the Cable Spreading Room and the Reactor Building. Note the multi-tiered trays, the dry chemical extinguishment (white), the tray corrosion, the exposed conductors and the scaffolding.

This is another close-up view of the five high, multi-tiered cable trays passing through the Reactor Building wall. Observe the cable damage,-soot on the ceiling and acid marking on the conduit.

Taken from the floor, this photo shows a four high tray array approximately 15–20 feet from the cable penetration. Imagine the difficulties encountered putting out the fires in these trays some 20–25 feet overhead.

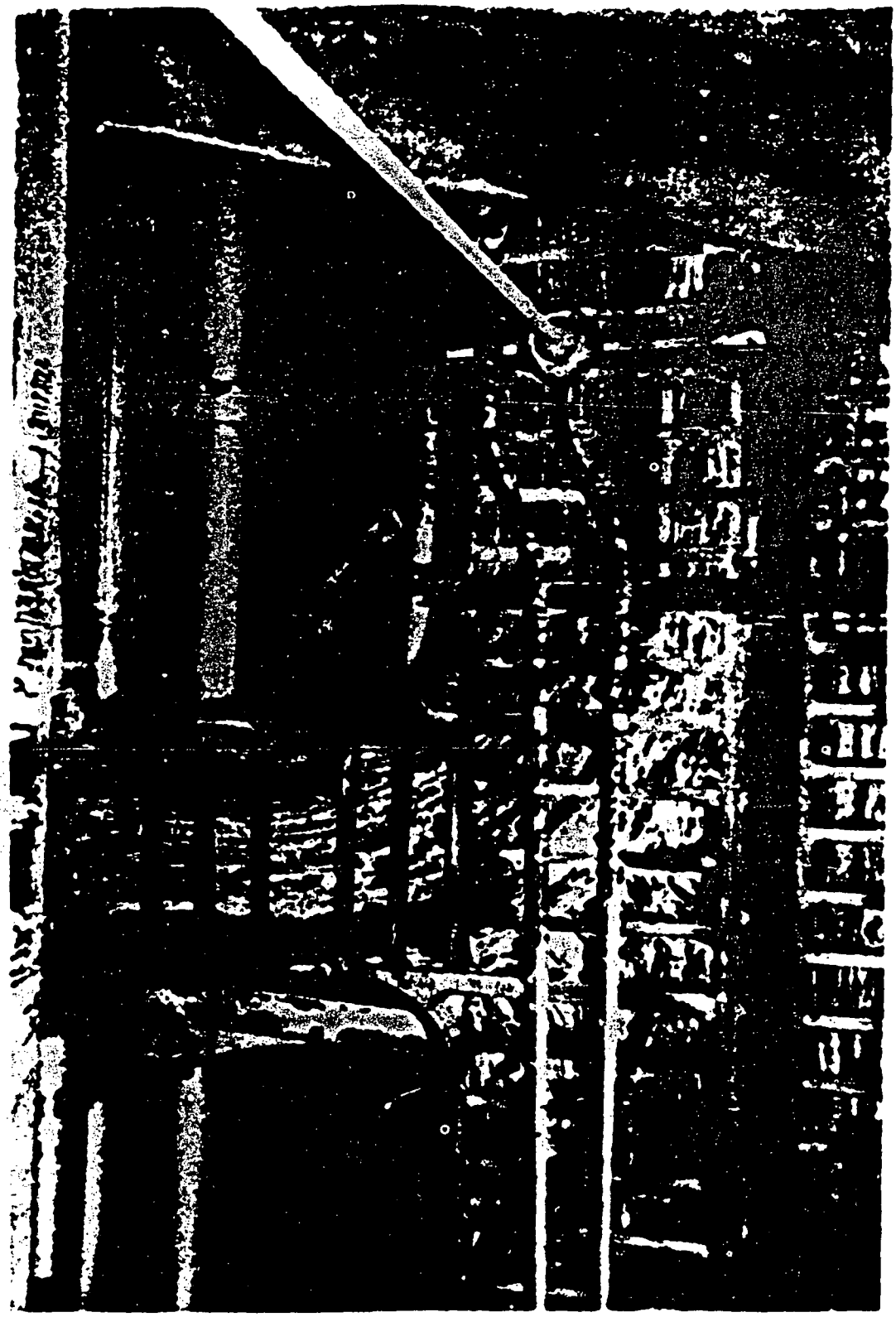


This is a close-up view of Photo No. 6. Observe the tray corrosion (a bluish-green color), the carbon deposits on the walls and ceiling and the exposed conductors in all four trays.



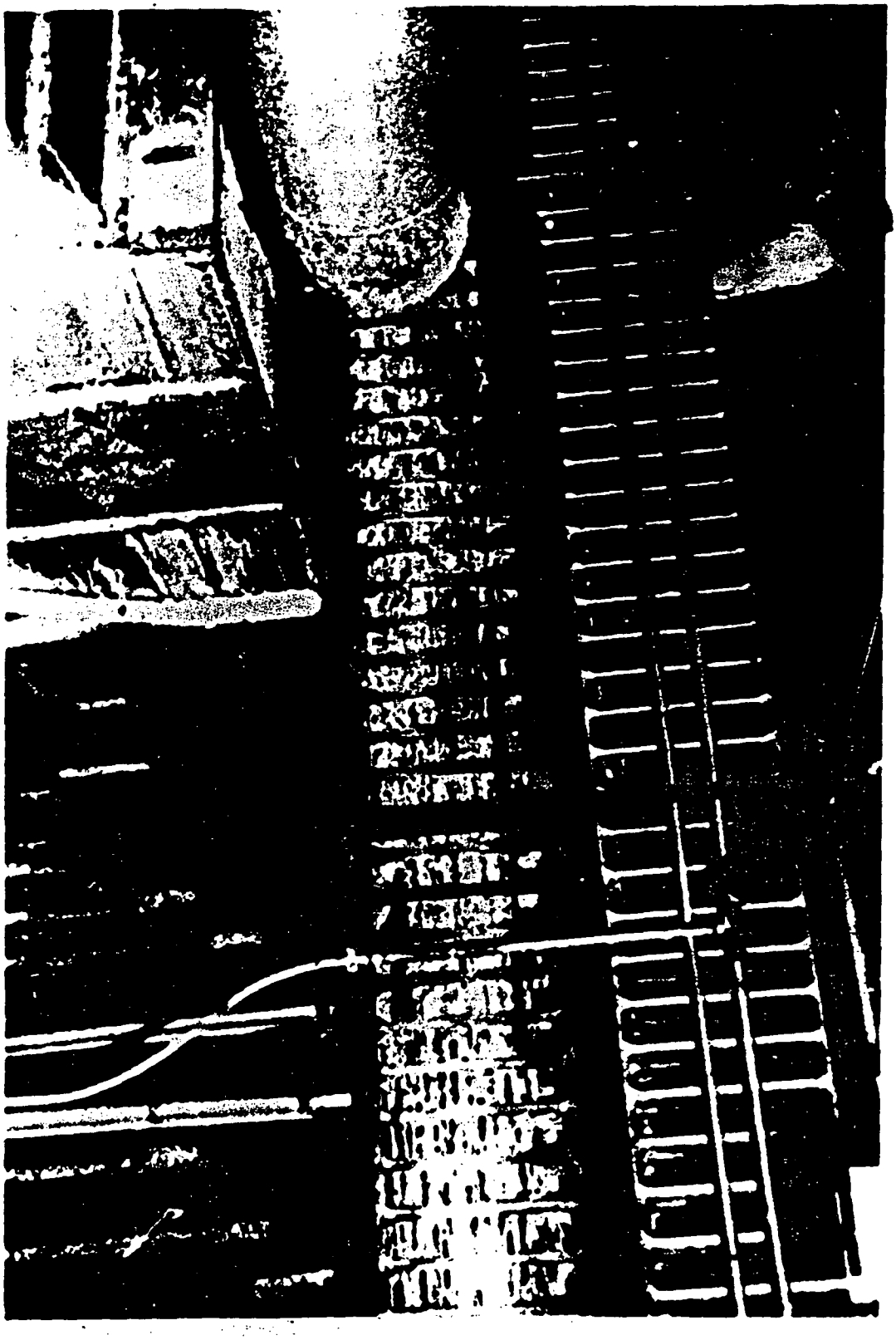
This is another view of the cable trays in Photos No. 6 and 7 but taken from the other side.

This photo shows the cable trays in Photos No. 6–8 passing through the wall directly opposite the Reactor Building-Cable Spreading Room wall. It was reported the fire penetrated this wall and continued another 3–4 feet.

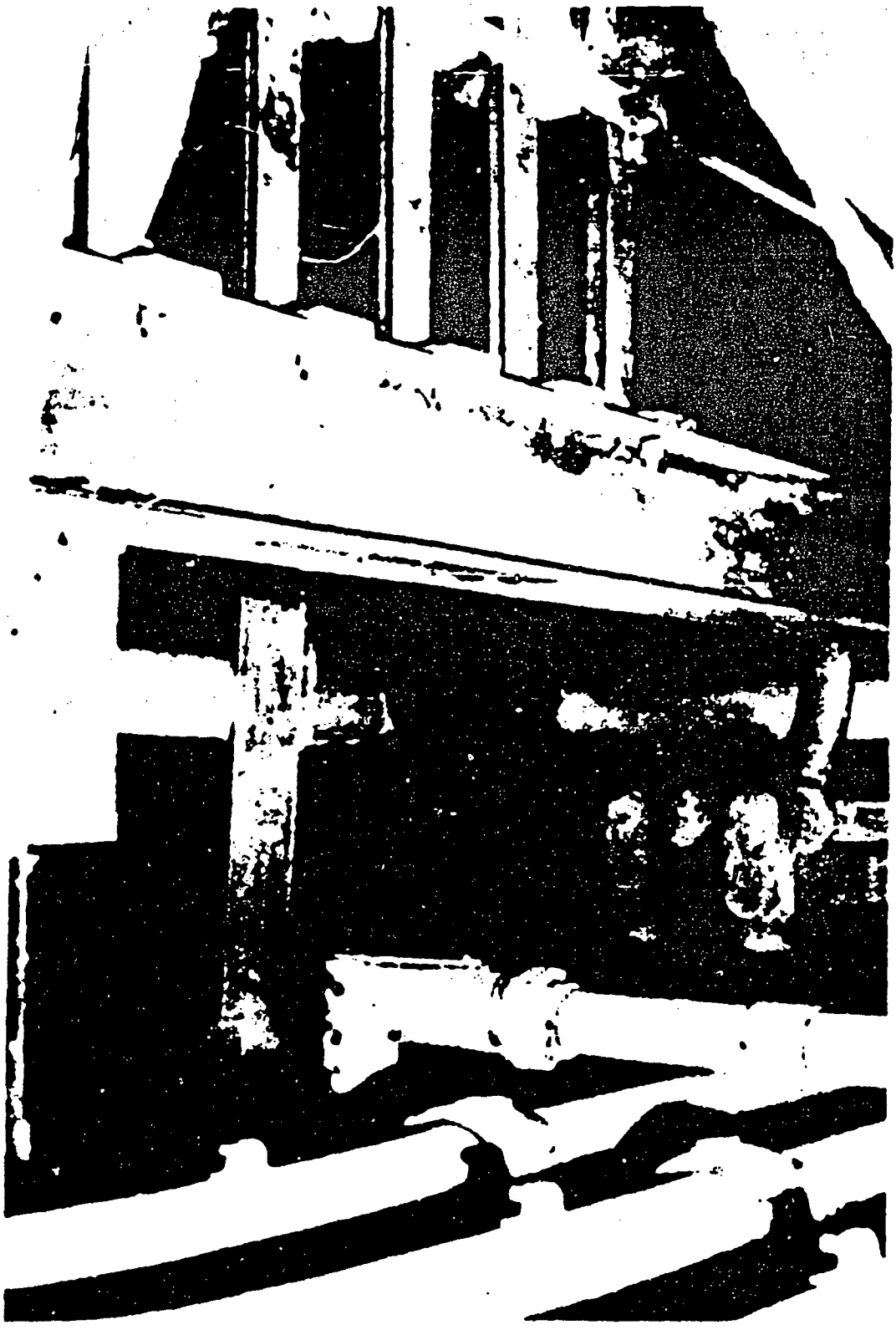


Looking up, this photo depicts some of the horizontal cable trays approximately ten feet from the cable tray penetration. Observe the exposed conductors, the thermocouple cable in the bottom tray, the dry chemical powders, the scaffolding and the close proximity of the parellel runs of vertically tiered trays. Apparently, redundant circuits were located in the upper cable tray.

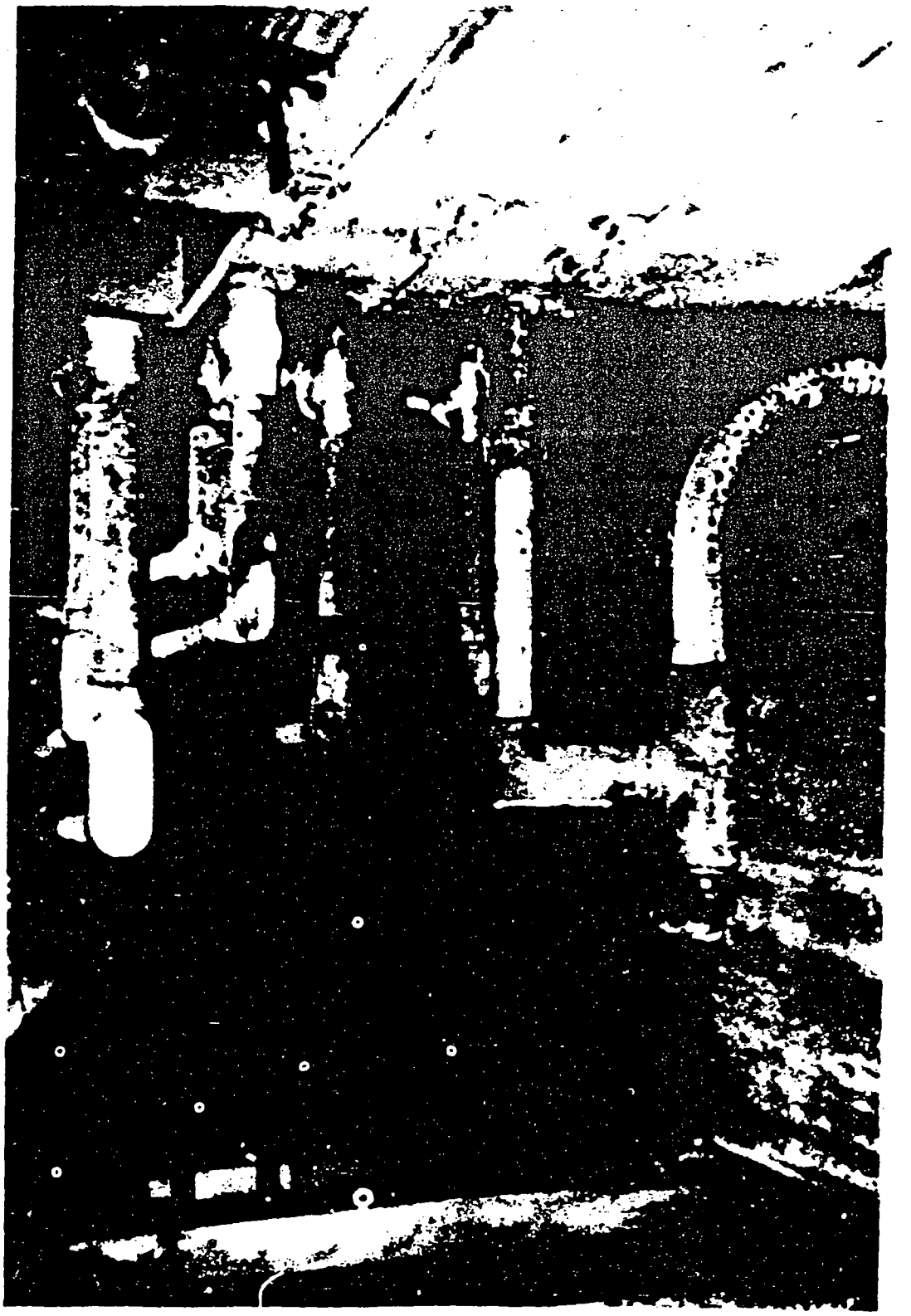
Once again looking up is a view of the horizontal cable trays approximately 15–20 feet away from the 4' x 4' cable tray penetration. Note where the fire stopped in the vertical trays in the center of this photo whereas in the top right corner tray, the fire continued downward 7–8 feet.



This photo illustrates how far the cable fire spread (35–40') in the direction parallel to the Reactor Building-Cable Spreading Room wall. Note the damage to the cable in the adjacent cable tray (upper), the dry chemical powders in the bottom tray, the acid markings on the conduit and the carbon deposits on the ceiling.



The cable fire also affected a 4160 volt power supply mounted inside an aluminum conduit. Either the heat of the fire or electrical arcing melted this portion of electrical conduit, exposing the conductors.



This is a close-up view of some electrical conduit penetrating the Reactor Building-Cable Spreading Room wall. Note the effects of corrosion on the conduit.

APPENDIX 9

QUESTIONS AND ANSWERS: JCAE, NRC

LETTER DATED SEPTEMBER 18, 1975, MURPHY TO KAMMERER, FORWARDING ADDITIONAL QUESTIONS TO NRC AND NRC RESPONSE.

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 18, 1975.

MR. CARLTON KAMMERER,
Director, Office of Congressional Liaison, Nuclear Regulatory Commission, Washington, D.C.

DEAR MR. KAMMERER: Enclosed is a list of additional questions relative to the September 16, 1975 NRC testimony on the Browns Ferry fire which the Committee would like you to provide answers to for the record. We also wish to receive your responses to several other matters raised during the hearing and for which it was agreed that material would be provided for the record. These include, but are not limited to, (a) a listing of the inspections that had been conducted on the Browns Ferry cable room and a copy of the most recent inspection report, (b) an analysis of agreements and disagreements between the NRC investigation report and the report prepared by the Nuclear Energy Liability and Property Insurance Association (NELPIA), (c) material addressing the probability of the Browns Ferry accident from the Rasmussen analysis viewpoint, and (d) an analysis of the paperwork required as a result of the accident, with specific regard to the NRC investigatory process. Please refer to the transcript of the hearings (to be provided to you within the next day or two) to determine the additional responses required.

Since we are endeavoring to publish and distribute the hearing record as rapidly as possible, it would be appreciated if you would provide your answers to the enclosed questions and the other responses no later than October 1, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, JR.,
Executive Director.

Enclosure.

QUESTIONS ON NRC TESTIMONY

Browns Ferry Hearing—September 16, 1975

1. On page 4 of Dr. Knuth's testimony, it is stated that use of an open flame to test for air leakage has been a relatively common practice in the utility industry. Was NRC aware of this practice before the fire, and had it been approved?

2. On page 14 of Dr. Knuth's testimony, it is noted that TVA has taken exception to some of the findings in NRC's investigation report, and that NRC is evaluating the validity of TVA's rebuttal. Does NRC have any conclusions yet on whether or not TVA's positions are valid?

3. On page 15 of Dr. Knuth's testimony, it is indicated that the use of punitive sanctions against TVA is not warranted. Could you describe in further detail the basis on which NRC reached this conclusion, considering the many deficiencies noted in its investigation report and the serious situation that resulted?

4. Apparently, the installation of the cable tray penetration sealant and fire retardant coating had not been completed prior to operations of Units 1 and 2 (Investigation Report—I-2). How did this manage to slip past detection by both TVA and NRC?

5. The procedures for notifying States and local support agencies left something to be desired. How closely had NRC reviewed these procedures?

6. How does NRC reconcile the need for water deluge systems in cable rooms (page 7 of Dr. Hanauer's testimony) with TVA's contention that the use of water might have led to further loss of control of the reactors?

7. The Committee understands that a petition has been filed to change the NRC regulations to require plant operators to distribute emergency plans to the public and conduct yearly drills. What is the status of NRC's consideration of this matter?

8. Over the years, the primary emphasis in nuclear safety has been placed on hypothetical nuclear accidents, such as loss of coolant accidents and reactor excursions. The fire at Browns Ferry and other fires over the years would appear to suggest that the accidents that have occurred at nuclear plants have been more conventional in nature. In this regard:

(a) What consideration does NRC give to fires and other conventional accidents in its safety reviews?

(b) How is NRC organized to assure such hazards are recognized; i.e., does NRC have any special expertise in fire detection and prevention?

(c) Does NRC intend to make any organizational changes to put increased emphasis on preventing fires and other conventional hazards?

9. Mr. David Comey has made two specific recommendations to NRC as a result of the fire:

(a) That operation of nuclear units not be permitted at stations where additional plants are under construction.

(b) That all reactors not fully complying with IEEE standards on separation of electrical cables be shut down.

What is the status of NRC's actions on these recommendations?

10. What is NRC's position regarding the continuing applicability of technical specifications in the event some departure from them is required as the result of an on-site emergency?

11. Please provide background on why the polyurethane material of the type tested by the applicant was not the type used. Also, discuss why these tests were relied upon, when it has been documented that small scale laboratory tests are inadequate to determine the fire properties of a material in large-scale applications. (AEC Serious Accidents Bulletin No. 336, September 19, 1974). Finally, why was polyurethane foam approved for use in such an inaccessible location, when the fire hazards of this practice have long been known? (AEC Health and Safety Bulletin No. 170, dated September 10, 1963 recommends that consideration be given to attaching polyurethane so that it can be removed more easily in case of fire).

[Nuclear Regulatory Commission response, subsequently received, follows:]

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., October 3, 1975.

Mr. GEORGE F. MURPHY, Jr.,
Executive Director, Joint Committee on Atomic Energy, Congress of the United States.

DEAR MR. MURPHY: Enclosed for inclusion in the record of the September 16, 1975 hearing of the Joint Committee on the Browns Ferry Nuclear Plant fire are the following:

1. Responses to requests for supplemental information made during the hearing.

2. Responses to the eleven questions transmitted to the NRC with your letter of September 18, 1975, to the Director, Office of Congressional Affairs.

Your letter called attention to four matters as having been raised during the hearing on which material should be supplied for the record, namely, (a) list of inspections of the cable room and a copy of the latest inspection report, (b) an analysis of the Nuclear Energy Liability and Property Insurance Association's report and NRC comments thereon, (c) material addressing the probability of the Browns Ferry fire from the Rasmussen analysis viewpoint, and (d) an analysis of the paperwork resulting from the occurrence.

Items (a), (b), and (d) are included in the responses contained in Enclosure 1. With respect to your request for additional material regarding the probability of the Browns Ferry fire—item (c)—the testimony presented at the hearing by Mr. Saul Levine (transcript pages 112-114) reflects all the information presently available on this matter. The analyses that formed the basis for the statement

at the hearing are still preliminary, and are being finalized in preparation for the publication of the final report on the Reactor Safety Study. The material requested concerning Browns Ferry is expected to be developed in about a month, and will be furnished to the Joint Committee when completed.

Sincerely,

BEN C. RUSCHE,
(For Lee V. Gossick),
Executive Director for Operations.

Enclosures:
(1) Items Requested for the Record.[1]
(2) Questions on NRC Testimony.

1. On page 4 of Dr. Knuth's testimony, it is stated that use of an open flame to test for air leakage has been a relatively common practice in the utility industry. Was NRC aware of this practice before the fire, and had it been approved?

Response: As stated in Dr. Knuth's testimony, the use of an open flame to test for air leakage in checking condenser vacuum has been a practice in the utility industry. The condensers are generally not located in areas containing vital safety equipment. Responses to IE Bulletins following the Browns Ferry fire indicate that open flames had been used for checking penetration leakage in areas containing vital safety equipment at three other reactor sites.

To the extent that many people on the NRC staff have had experience in the utility industry, the practice of using an open flame to check for condenser leaks was known to exist. The practice was neither explicitly approved nor disapproved by NRC in the condenser application. The more usual practice in industry, as indicated in the responses to our bulletins following this fire, is to administratively control the use of open flames in areas containing vital safety equipment by means of an "open flame" work permit.

2. On page 14 of Dr. Knuth's testimony, it is noted that TVA has taken exception to some of the findings in NRC's investigation report, and that NRC is evaluating the validity of TVA's rebuttal. Does NRC have any conclusions yet on whether or not TVA's positions are valid?

Response: Since the JCAE hearing on the Browns Ferry fire, the I&E Staff completed it evaluation of TVA's response to the NRC Notice of Violation. A copy of its letter to TVA dated September 29, 1975, on this matter is enclosed. The accident investigation report, TVA's response, and the I&E Staff evaluation of the TVA response will all be reviewed by the Special Review Group and by the Commission.

NUCLEAR REGULATORY COMMISSION,
REGION II,
Atlanta, Ga., September 29, 1975.

Attn.: Mr. J. E. Watson
TENNESSEE VALLEY AUTHORITY,
Manager of Power,
Power Building,
Chattanooga, Tenn.

GENTLEMEN: This refers to your September 2, 1975 response to the Notice of Violation sent to you by this office on July 28, 1975.

Of the items of apparent noncompliance in Appendix A of our July 28, 1975 Letter, we have no further questions on Items 1, 2a(1), 2a(2), 3a, 3b, and 3c. Appendix A to this letter notes that item 3d(2) is withdrawn as an item of noncompliance and gives our position on your reply to Items 2a(3), 2a(4), and 3d(1). Appendix B of this letter supplies our position on certain of your replies to the Areas of Concern contained in our letter of July 28, 1975. We will be examining these matters during future inspections and no further reply is requested of you at this time.

Appendix C to this letter provides our comments to certain of your responses on the Conclusions and Summary of Findings of Facts sections of our investigation report.

If you have any questions regarding these matters, please contact me.

Sincerely,

NORMAN C. MOSELEY,
Regional Director.

Attachments:
Appendices A, B, C.

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[1] Included at appropriate point in record.
[2] See Appendix 14 for September 2, 1975 letter.

IE POSITIONS RELATIVE TO TVA RESPONSE TO CERTAIN ITEMS OF NONCOMPLIANCE IDENTIFIED IN THE LETTER OF SEPTEMBER 2, 1975, FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

APPENDIX A

Item 2.a(3)

The Browns Ferry Procedure for "*Fire—Explosions—Natural Disasters*," contains the following:

"Shift Engineer on Duty Will:

1. Proceed to scene and appraise situation (the Shift Engineer may choose to remain in the control area and delegate on-scene responsibility to an Assistant Shift Engineer) . . .

5. Remain at the scene and supervise activities."

A normal shift complement at Browns Ferry is one Shift Engineer who is in overall charge of the entire plant and at least one Assistant Shift Engineer for each unit. In the event of a fire the TVA procedure requires delegation of on-scene responsibility when the Shift Engineer is not at the scene to supervise activities. In this case the Shift Engineer departed from the scene without making such delegation. This constitutes noncompliance with TVA procedures, adherence to which is required by TVA technical specifications. Although the Shift Engineer may have thought the Assistant Shift Engineer would assume this responsibility, the TVA procedure is clear in requiring specific delegation.

Our position continues that failure to make a specific delegation of responsibility constitutes noncompliance.

Item 2.a(4)

The first sentence of the quotation from the Browns Ferry Standard Practices Manual BFS3 is "Plant fire protection systems shall be fully operational at all times." (emphasis added). At the time of the fire, the Cable Spreading Room CO_2 installed system was not fully operational because (1) the metal plates were installed under the breakout glass used in the mechanical valve actuation method and (2) the manual electric system was disabled on the Unit 1 side by a switch located on Unit 2 side. From the Unit 1 side manual electric is the only actuation method, therefore, at the time of the fire the CO_2 system could not be promptly activated at all from that side because the switch on the Unit 2 side had de-energized the control on the Unit 1 side. Accordingly, the CO_2 system could not be considered fully operational.

Our position continues that this item represents noncompliance since the system was not fully operational.

Item 3.d(1)

In amendment 25 to the FSAR, beginning on page R7.5–11, the following sentence appears: "The power cables from the 4160 volt shutdown boards to the respective transformers for the 480 volt shutdown boards and the 480 volt diesel auxiliary boards are installed in separate conduits."

Whatever the original design classification of the 4 Kv shutdown board cables was, it is clear that the quoted statement from Amendment 25 requires that they be in separate conduits. Accordingly, the as-installed condition is not in compliance with the FSAR as amended.

Item 3.d(2)

The table on page III–8 of our investigation report is in error in that 1C Core Spray and Residual Heat Removal Normal Valve power supply is 4 Kv SD Board A through NOV Board 1B. Therefore, we concur that Division I and II separation was not violated in the cases discussed in this item. This apparent item of noncompliance is withdrawn.

IE POSITIONS RELATIVE TO TVA RESPONSE TO CERTAIN AREAS OF CONCERN IDENTIFIED IN THE LETTER OF SEPTEMBER 2, 1975 FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

APPENDIX B

Item 4

"Plant operations personnel and Public Safety Services personnel at the plant were not familiar with the reactor building fire extinguishing equipment

in that they did not know that the nozzle on the hose was specifically designed for fighting electrical fires."

This concern is based in part on actions by Shift Engineer U and Assistant Shift Engineer Y. Both of these individuals participated in the first attempt to use water. The attempt was not successful and led to a decision to try the nozzle from the Athens Fire Truck. Had these individuals been knowledgeable of the characteristics of the nozzle on the hose, it is doubtful that they would have concluded that another nozzle would perform better. These individuals had received TVA fire training and have job responsibilities for fire fighting. These actions led to a conclusion that these key people did not demonstrate knowledge of the equipment available. Your September 2, 1975, letter stated that additional training has been accomplished. This area will be reviewed in future inspections.

Item 5

"Emergency breathing apparatus appears not to have been properly maintained."

This concern was developed from statements made by various personnel who were interviewed by the investigators. Excerpts from specific statements included in the investigation report are as follows:

(1) *ASE-NNNN.*—"The unit that PSO-CC had would not puncture the canister and I don't remember the problem with the other unit. But the Chem-Ox mask units did look to be in sad shape in my opinion."

(2) *IE-JJJJ.*—"I heard that the air supplies were hard to find and did not fit well together."

(3) *AUO-Q.*—"About half the airpacks were unusable because they were marked 'not charged' or 'no mask'." In an explanatory note, AUO-Q stated, "This is very misleading. The airpacks that were brought to the fire site by some of the maintenance people were unuseable due to marking. They simply failed to notice the markings. We had enough airpacks at the site but a lot of the airpacks brought to the fire site were not charged or had the masks in them."

These statements coupled with additional statements concerning lack of available air in cylinders led to our concern that maintenance activities may not have been promptly performed on emergency equipment.

Item 7

"There was apparently no attempt by CECC management to obtain expert advice on methods of fighting the fire."

The plant superintendent, by his discussions with TVA management, gave evidence of his concern regarding his decision not to use water on the fire at an early stage. In view of the fact that local expert advice for earlier use of water—first given by the Athens Fire Chief at about 2:00 p.m. was not heeded—it appears that TVA management should have secured and provided expert advice to the plant superintendent on methods of fighting the fire. The investigation revealed no positive recommendation—nor definitive advice—by TVA management concerning use of water. It appears that such recommendations are an assigned function of the off-site management control in an emergency such as this fire.

We continue to believe that earlier use of water to extinguish the fire would have reduced the impact of the fire on your plants. The increasing inoperability of equipment as the fire progressed, in our view, supports the position that earlier positive extinguishing of the fire by water was called for. The water nozzle at the scene was designed to permit its use with minimum hazard to personnel.

Item 10

"There was apparently no attempt by the CECC Director to direct the operations of the center. The plant superintendent was involved in extensive communications with the center rather than directing the fire fighting and recovery operations. As a consequence the efforts at both locations were not coordinated, management personnel were not provided with accurate information and their actions were relatively ineffective."

The information gained during the IE investigation indicates that the control centers served more as communication centers than as centers to provide an overseeing management control of the situation. Particularly difficult to identify were examples where the control centers evaluated and made recommendations concerning the situation and actions taken to assure that support efforts were adequate.

We could identify no control center review of the measures being used to control the emergency. The control centers appeared to be deeply involved in the hour-by-hour unfolding of the situation rather than in providing the technical and managerial perspective which were needed for a more controlled response to the situation.

Item 11

"Although plant procedures specify that the unit operator has the authority to shutdown the reactor in the event of an emergency, his responsibility to effect a shutdown promptly has not been defined by management."

Your response states that the operator "may operate on his own initiative" (emphasis added) to take actions in emergencies where there is not time to consult with supervision. This position, more positively stated, should be emphasized to your operators as the TVA management position. We believe that the operator has the positive responsibility—shall operate—to take actions on his own initiative when significant, unexplained anomalies occur and he does not have time to consult with supervision.

IE COMMENTS TO TVA'S RESPONSE CONCERNING CONCLUSIONS AND SUMMARY OF FINDINGS OF FACTS IN THE LETTER OF SEPTEMBER 2, 1975, FROM J. E. GILLELAND, TVA, TO NORMAN C. MOSELEY, NRC

Since the Items of Noncompliance (Appendix A) and Areas of Concern (Appendix B) are founded on the Conclusions and Summary of Findings of Facts sections of the investigation report, those comments in Appendix C deal only with matters not already addressed in Appendix A and Appendix B.

APPENDIX C

CONCLUSIONS

B.1.(e)

"Inadequate training of TVA personnel in fire fighting procedures and equipment."

This conclusion is based on the overall response of TVA in fire fighting founded on statements made by individuals during the IE investigation. Some individuals stated they had received no fire fighting training; others said that the training they had received was some years prior to the fire.

B.2.(b)

"Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1."

Collective finding of the IE investigating team indicates that there was a lack of coordination of shutdown and restoration activities until after the initial manual depressurization of Unit 1. Though TVA does not agree with our conclusion regarding the extent of direct supervisory involvement in the decision to initiate the blowdown of Unit 1, it is based on the investigators' discussions with those involved and rests heavily on the impression that the operator did not believe he had been directed to initiate the blowdown. In any case, the plant superintendent's involvement in coordinating shutdown activities during the time in question could not have been extensive since he arrived onsite between approximately 1:20 p.m. and 1:35 p.m. The initiation of manual blowdown of Unit 1 occurred at approximately 1:40 p.m. The conclusion, therefore, is based primarily on the actions of supervision on the site during the entire period in question.

B.2.(c)

"Failure to establish priorities for restoration of equipment. This resulted in uncoordinated, and in some cases, counter productive or duplicate attempts to restore key equipment to service."

This conclusion is related to activities prior to extinguishing the fire. Our investigation revealed no evidence of a systematic approach to establishing response action alternatives and the associated priorities during that period.

FINDINGS OF FACTS

A.5

The location of the fire barrier in the penetration did not conform to the approved design drawings.

We concur with your contention that the location of the fire barrier in the penetration would have no bearing on the function of the fire barrier if the installation had been completed with the flameproof coating. As discussed, however, in the conclusion section of the report (IE Inspection Report, Summary Section B.1.(h)), the inaccessibility of the initial ignition location due to the position of the penetration contributed to the cause of the fire and to its severity. If the penetration had been located ten inches closer to the spreading room wall as specified in the design drawing, the likelihood of the fire would have been decreased and initial efforts at extinguishing the fire would have been aided through improved accessibility.

A.14

"Contrary to good safety practices, the plant procedures do not restrict the use of elevators during fires; both operations and construction personnel used the plant elevators while the fire was in progress."

In the event of the failure of an elevator during a fire, any passengers would be exposed to hazards created by that fire until rescued. Although no one was trapped, at least one elevator used by Browns Ferry personnel failed during the fire.

D.1

"The Athens Fire Department (AFD) responded promptly to the notification of the fire and was prepared to assist within approximately 35 minutes of receiving the alarm. Athens is about ten miles from the site and, upon arrival, the firemen had to be issued personnel radiation monitoring devices prior to entry."

The information concerning issuance of radiation monitoring devices to firement prior to plant entry was included to explain the 35 minute time interval, not to criticize the practice. We do believe that attention should be given to minimizing delays in emergency response while maintaining due attention to safety and security concerns.

D.6

"The Director of the CECC was not aware that the fire had been extinguished until approximately 8:45 p.m. when in fact the fire was extinguished prior to 7:45 p.m."

We do not agree that the Director of the CECC was aware at about 7:45 p.m. CDT that the fire had been extinguished as you maintain. The CECC Director's log and that of individual CECC-AF, a co-worker, indicate that at about 8:05 p.m. CDT the State of Tennessee was advised of the fire. The State of Tennessee records indicate that they were advised of the fire at 8:15 p.m. CDT. Neither the Director nor CECC-AF, therefore, were aware that the fire was out at about 8:05 p.m. CDT.

Subsequent to the call to Tennessee, an untimed late entry was made which stated: "(no time) 8:45 p.m. fire (reported) out." The next entry was made at 10:30 p.m. CDT. We concur with you that it is conceivable that the time in the Director's log could be 8:45 p.m. EDT rather than CDT. This does not establish that the Director was aware at 7:45 p.m. CDT that the fire was out since the entry was made subsequent to 8:05 p.m. CDT. We conclude that the Director became aware that the fire was out between about 8:15 p.m. and 10:30 p.m. CDT, the time of the next log entry. The terminology of the untimed entry indicates that it was at 8:45 p.m. when the event was reported to him.

D.7

"Communications by the CECC with State and local agencies focused on plant operating status rather than offsite radiological releases which is the prime responsibility of these agencies."

The results of the investigation including interviews with personnel from the states and reviews of their records indicate that much of the information provided by TVA directly or in response to questions was in the nature of "raw"

operational details rather than specific information which would facilitate evaluation by the states of the probability and nature of possible offsite radiological releases.

We did not intend to imply that *only* offsite radiological release information is to be provided to the states. The states do need to know plant operational status in order to plan their activities in their jurisdiction.

3. On page 15 of Dr. Knuth's testimony, it is indicated that the use of punitive sanctions against TVA is not warranted. Could you describe in further detail the basis on which NRC reached this conclusion, considering the many deficiencies noted in its investigation report and the serious situation that resulted?

Response: The primary emphasis of the NRC's enforcement program has been to bring about corrective action on the part of violators. Punitive action has not been a primary objective of the enforcement program. This philosophy is set forth in a letter dated December 31, 1974 from Dr. Donald F. Knuth to all licensees transmitting the criteria used by the Staff in determining the appropriate enforcement for any given situation. Copies of the December 31, 1974 letter and the enforcement criteria are enclosed.

NRC formal enforcement actions carried out by the Office of Inspection and Enforcement are taken in accordance with IE Manual Chapter identified as MC-0800 which is part of the public record. Under the enforcement criteria, the action decided upon in a given case depends essentially on three factors:

The significance of each time of noncompliance.
The total number of items of noncompliance, and
The enforcement history of the licesnee involved.

These factors require the exercise of judgment in their application to specific situations. To assist in guiding this judgment general criteria have been established. These criteria categorize items of noncompliance into three levels of significance as follows: violations, infractions and deficiencies.

The most significant items of noncompliance are denominated "violations". Violations are defined in terms of items of noncompliance which would lead to or result in relatively severe types of incidents or occurrences such as, for example, a radiation exposure exceeding 50 times the established regulatory limits.

The next most significant items of noncompliance are denominated "infractions." Infractions" are defined in terms of several specific types of incidents or occurrences of a less severe nature than those amounting to "violations." For example, a radiation exposure merely exceeding established regulatory limits would normally be an "infraction."

The least significant item of noncompliance is defined as a "deficiency." Deficiencies are minor matters such as failures to comply with records, posting, or labeling requirements.

In addition to the categorization of items of noncompliance into levels of significance, further guidance in the exercise of the judgment as to the appropriate enforcement action in any given case is set forth in the Office of Inspection and Enforcement Manual. There a point system is established to guide the determination of what enforcement action is appropriate in a given situation.

Basically, three levels of enforcement action are available to the NRC in the discharge of its enforcement responsibilities:

Notice of violation.
Civil penalty.
Order modifying, suspending or revoking a license or to cease and desist from an unauthorized activity.

The selection of the appropriate enforcement action, as indicated above, is made after consideration of both the significance of the individual items of noncompliance as well as the accumulated number of points assigned to the total noncompliance items. Thus, for example, under our criteria, a single serious item of noncompliance could result in a civil penalty or even an order modifying, suspending, or revoking a license. On the other hand, a whole series of minor items of noncompliance might not result in a sufficient number of points to warrant anything more than a notice of violation under our criteria.

Of course, the point system is merely guidance. Similarly, the important determination as to whether an item of noncompliance falls into this category or that category can never be completely automatic. Judgment is always involved. Indeed in the definition of "violation" there is a catch-all category:

"Other similar items of noncompliance having actual or potential consequences of the same maugnitude."

Each of the items of noncompliance in the Browns Ferry case was evaluated under our criteria as to its level of significance. None of the items of noncompliance could be reasonably considered as a "violation" since none of the five QA items nor the one technical specification item can fairly be said to have caused, contributed to or aggravated (or to have had substantial potential for so doing) an incident or occurrence of the type set forth in our criteria defining "violations." In view of the catch-all category "other similar items of noncompliance having actual or potential consequences of the same magnitude," the position could to taken that inadequate Q/A contributed to a serious fire and that a serious fire is a "consequence of the same magnitude," as, for example, radiation levels exceeding 50 times established limits. However, the fire "resulted in a reduction of preventive capability below requirements but redundant controls precluded an item of noncompliance of the violation category"—an explicit definition of an "infraction." Consequently, the individual items of noncompliance all are categorized as infractions-redundancy was reduced but still existed. Evaluating the six "infractions" revealed by the investigation against our guidance did not result in the number of "points" normally associated with a civil penalty.

Based on this evaluation, a civil penalty was not considered to be the appropriate enforcement action because no single item of noncompliance reached the significance of a violation nor did the aggregate of the items of noncompliance reach the significance of a violation.

U.S. Atomic Energy Commission,
Washington, D.C., December 31, 1974.

To: All AEC Licensees.

Criteria for Determining Enforcement Action and Categories of Noncompliance With AEC Regulatory Requirements—Modifications

On November 1, 1972, the Commission issued criteria for enforcement actions to be taken for noncompliance with its rules and with license conditions in accordance with Sections 161, 186, and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR. On June 5, 1973, the Commission notified licensees that categories of violation with AEC regulatory requirements had been established because the Commission and the nuclear industry recognized that the significance of violations varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment.

Based on a review of the experience with the criteria for determining enforcement action and the categories of noncompliance, modifications of the use of these criteria and these categories are being made. Comments explaining the modifications are enclosed as Attachments A and B.

The changes in the criteria and categories are primarily administrative in nature and should result in a higher level of understanding of the enforcement program—and the results of the program—on the part of the public and the industry. The basic purpose of the enforcement program—enhancement of the health and safety of the public, the common defense and security, and the environment—remains the same. The long standing practice of requiring corrective action for each identified item of noncompliance (Violations) is not changed. The enforcement program continues to emphasize corrective action where necessary to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment.

The modifications clarify the enforcement criteria and categories of noncompliance in the areas of safeguards and environmental matters and provide more explicit definitions to aid in a better understanding of the enforcement program. These definitions make clear the applicability of the program in matters of

quality assurance, management control, and systems performance. Also, because the Commission relies to a degree on reports from licensees to assure that timely corrective action is taken and to assure that the industry is notified of important matters of generic interest, a reporting requirement is viewed from the enforcement standpoint to be of the same level of importance as the matter for which the report is required. As a part of the correspondence between a licensee and the AEC subsequent to an inspection, notifications will be made to a licensee of apparent failures on the part of the licensee to meet his commitments contained in his application or in correspondence to the AEC and of deviations from appropriate codes, standards or guides.

The levels of enforcement actions available to the Commission in the exercise of its regulatory responsibilities are the same as those set forth in the letter of November 1, 1972. These include written notices of violation, civil monetary penalties, and orders to "cease and desist" or for modification, suspension, or revocation of a license.

The criteria for issuance of a "Notice of Violation" are essentially unchanged.

The criteria for civil penalties have been modified to elaborate upon those situations for which civil penalties may be imposed. The amount of civil penalty in any given case, within the confines of the amounts established by the Atomic Energy Act, is determined by consideration of several factors including:

1. Potential or actual consequences associated with the item of noncompliance. This includes consideration of the categories of noncompliance.

2. Type of licensee. This includes the purpose for which licensed and the quantity, form and kind of radioactive material authorized.

3. The licensee's recent enforcement history, if applicable. This includes the nature and number of items of noncompliance, the frequency of noncompliance, whether items of noncompliance were repetitive of the same or similar requirements, promptness of corrective action, and the licensee's management of its program for assuring compliance with regulatory requirements.

The criteria clarify that repetitiveness of noncompliance or history of noncompliance is not an essential ingredient for consideration for civil penalty. In some cases of a single instance of noncompliance, a civil penalty may be the appropriate enforcement sanction.

The criteria for orders emphasize the importance of quality assurance and are broadened to include all aspects of the regulatory program. Under these criteria, an order to suspend a license or a portion thereof may be issued for authorized activities of licensees or permit holders which are performed in such a manner as to constitute an immediate or potential threat to employees or the public; or for construction deficiencies which, if not suspended immediately, could eventually result in significant or essentially irreversible construction defects which impact on safety or which increase the potential for or the potential severity of an accident. If, for example, a quality assurance requirement for a specific construction activity is not implemented, this activity may be suspended until full compliance with the requirement is achieved.

Regulatory Operations Bulletins and Immediate Action Letters have been used not only to disseminate information but also as a means of accomplishing voluntary action on the part of licensees to inspect, report and make commitments to correct problems on a timely schedule. These two communications are recognized in these revisions. If these methods are ineffective in achieving the desired action, an order may be promptly issued requiring the action.

The enforcement record of a licensee may be a consideration in selecting the appropriate enforcement sanction in any given case. A licensee's enforcement history is evaluated in terms of distribution of items of noncompliance by importance and by the degree of repetitiveness of noncompliance with the same basic requirement. However, regardless of the history, consideration will be given to the more significant enforcement sanctions as a result of any inspection that reveals items of particular importance to safety and management.

The former system of severity categorization, which was the subject of a letter to licensees dated June 5, 1973, has been revised to place items of noncompliance with regulatory requirements (Violations) more clearly in perspective with regard to their relative significance to the public health, safety and interest and

the common defense and security. As shown in Attachment B to this letter, the revised system for categorizing violations (items of noncompliance) has three levels of relative importance which are designated in descending order as (1) "violation," (2) "infraction," and (3) "deficiency," each of which is a legal violation in the statutory sense.

It should be recognized that the enforcement criteria and the categories of noncompliance apply only to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements. The licensee may also be notified of deviations from commitments and appropriate codes, standards, or guides. The significance of these failures generally is judged against the actual or potential consequences resulting from the failures and from the standpoint of licensee awareness and management of his program. From the viewpoint of enforcement, a licensee failure that results in the potential for consequences is equally important with the failure that results in the consequences—both represent instances of failure of the licensee to properly perform. However, from the impact of health and safety, common defense and security, the protection of the environment, actual consequences—when the event did occur—and potential consequences—when the opportunity for occurrences exists but the event did not happen—of a item of noncompliance are quite different. In reporting the more important items of noncompliance, those items that caused or resulted in actual consequences will be differentiated from those that merely provided the potential for the consequences.

The enforcement criteria and the categories of noncompliance apply to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements, commitments, and appropriate codes, standards or guides. There do occur events—such as some equipment malfunctions—at licensee facilities which are not founded in the failure of the licensee to meet requirements, commitments, and appropriate codes, standards, and guides. Such events are not included within the enforcement program.

The enforcement criteria and the categories of noncompliance have been placed in the Public Document Room, 1717 H Street, N.W., Washington, D.C., and a notice has been placed in the Federal Register concerning their availability to all persons upon request.

Sincerely,

DONALD F. KNUTH,
Director of Regulatory Operations.

Enclosures:
A. Criteria for Determining Enforcement Action.
B. Categories of Items of Noncompliance.

CRITERIA FOR DETERMINING ENFORCEMENT ACTION

IN CONNECTION WITH LICENSING AND REGULATORY PROVISIONS OF THE ATOMIC ENERGY ACT OF 1954, AS AMENDED, AND REGULATIONS AND LICENSES ISSUED THEREUNDER

Introduction

The purpose of the AEC enforcement program is the enhancement of the health and safety of the public, the common defense and security, and the environment. The enforcement program emphasizes corrective action, where necessary, to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment. Corrective action is required for each identified item of noncompliance.

Results of AEC inspections and investigations of licensed activities have shown that licensees have not in all cases complied with the regulatory requirements, and it has been necessary to take specific enforcement actions commensurate with the items of noncompliance. This document sets out the criteria for enforcement actions to be taken with respect to future noncompliance with the Atomic Energy Commission's requirements in accordance with Sections 161, 186 and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR.

Levels of enforcement actions available to the Commission

The formal actions available to the Commission in the exercise of its enforcement responsibilities are of three basic types (notices of violation, civil penalties, and orders) which may be applicable to a specific enforcement situation.

1. Written Notices of Violation (10 CFR 2.201).

Notices of Violations are written notices to licensees, citing the apparent instances of failure to comply with regulatory requirements (Violations) which for purposes of categorization have been classified violations, infractions and deficiencies. Such items of noncompliance are generally observed or identified during investigations, inspections, or inquiries.

The same letter enclosing a Notice of Violation may also enclose a notification of apparent deviations from licensee commitments and the provisions of appropriate codes, standards or guides.

2. *Civil Monetary Penalties (10 CFR 2.205).*

The Commission may levy civil monetary penalties against licensees for violations, infractions or deficiencies with respect to requirements in licensing provisions of the Act or any rule, regulation, order, or license issued thereunder. The Commission is required to issue a "notice of violation" to the person charged before instituting proceedings to impose a civil penalty.

3. *Orders to Cease and Desist; and Orders for Suspension, Modification, or Revocation of a License (10 CFR 2.202 and 2.204).*

The AEC has authority to issue orders to "cease and desist," and orders to suspend, modify, or revoke licenses. Such orders are ordinarily preceded by certain procedural requirements, including a written "notice of violation" to the licensee providing him with an opportunity to respond as to the corrective measures being taken. In the event the licensee fails to respond to the notice or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In some instances where the health, safety, or interest of employees or the public so requires or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

In addition to proceeding by way of order, the Commission may also, pursuant to Section 232 of the Act, request the Attorney General to obtain an injunction or other court order to enjoin licensees from violating the Act or any regulation or order issued thereunder.

NOTICE OF VIOLATION—CRITERIA

Section 2.201 of 10 CFR requires that before any formal enforcement action is taken for alleged noncompliance, the AEC will serve on the licensee a written "notice of violation" except when the Director of Regulation finds that the public health, safety, or interest so requires, or that noncompliance is deliberate, the "notice of violation" may be omitted and an order to show cause issued.

Generally, a "notice of violation" may be considered sufficient enforcement action in those cases where:

(*a*) Items of noncompliance are readily correctable, or

(*b*) Items of noncompliance are not repetitive or numerous, and do not constitute an immediate or serious threat to the health and safety of the licensee's employees or the public, to the environment, or to the common defense and security, and

(*c*) There is no indication that appropriate corrective action will not be taken.

CIVIL MONETARY PENALTIES—CRITERIA

The Commission may levy civil monetary penalties on licensees who do not comply with the licensing provisions of the Act or any rule, regulation, order, or license issued. Generally, the type of cases that are appriatepr for civil nplae or license issued. Generally, the type of cases that are appropriate for imposing civil penalties are those involving significant items of noncompliance and which represent a threat (but not necessarily immediate) to the health, safety, or interest of the public, or to the common defense or security, or the environment. As a matter of judgment, civil penalties may be used in lieu of license suspension when there is no immediate threat to the health and safety or the common defense and security and license suspension would deprive the licensee or his employees of their means of livelihood, or the public of essential service.

Civil penalties may be the appropriate enforcement action in cases or situations which meet one or more of the following criteria:

(*a*) Those cases of noncompliance with the same basic requirements that were brought to the attention of the licensee in a "notice of violation" following a previous inspection; or

(*b*) Those cases of noncompliance in which the licensee fails to carry out in a timely manner the corrective action the licensee stated would be taken in response to a previous written notice; or

(*c*) Those cases involving the *deliberate* failure of a person to comply with regulatory requirements;[1] or

(*d*) Those cases involving items of noncompliance in which (1) the licensee's history is one of chronic noncompliance, or (2) due to the nature and number of items of noncompliance, it is apparent that management, having been afforded an opportunity to correct previous items of noncompliance, is not conducting its licensed activities in conformance with regulatory requirements, or

(*e*) Those cases where (1) an order for immediate, but temporary, suspension or to "cease and desist" is issued to remove an immediate threat to the health or safety of the licensee's employees or to the public, to the environment or to the common defense and security, and (2) punitive action is deemed necessary to assure future compliance; or

(*f*) Those cases involving activities under construction permits where there are repeated items of noncompliance with regulatory requirements; or

(*g*) Those cases where an item of noncompliance resulted in or contributed to the cause or to the seriousness of an accident or an incident; or

(*h*) Those cases involving items of noncompliance in the Violation category; or

(*i*) Those cases where the nature and number of items of noncompliance with the regulatory requirements identified during an inspection or an investigation demonstrate that management is not conducting its licensed activities with adequate concern for the health, safety or interest of its employees or the public or the common defense and security; or

(*j*) Those cases where licensees knowingly use materials which are not authorized by the license or utilize authorized materials for uses which are not authorized; or

(*k*) Those cases where significant matters[2] were not reported to the Commission in a timely manner as required by the regulatory requirements.

Civil penalties may be assessed for other cases having comparable types of items of noncompliance and situations for which the Commission deems civil penalties to be appropriate and necessary.

ORDERS—CRITERIA

The AEC has authority to issue orders to "cease and desist" or to suspend, modify, or revoke licenses. The Commission is empowered to enforce these orders and obtain any other appropriate relief by injunction from Federal district courts, if necessary. Cases involving an immediate threat to the public health and safety, or the common defense and security, require immediate steps to remove the threat and are handled by this type of action. Persons who deliberately violate, attempt to violate, or conspire to violate the Commission's regulations and orders, are, upon conviction of the violations, subject to fine up to $5,000 and imprisonment for not more than two years (Section 223 of the Act).

In the event the licensee fails to respond to a "notice of violation" or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In those instances where the health, safety, or interest of employees or the public, or the common defense and security so requires, or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

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[1] NOTE: Section 221(b) of the Atomic Energy Act requires the FBI to investigate all suspected or alleged criminal violations of the Act.

[2] Such significant matters may include, but are not limited to, exposure of personnel to doses in excess of limits, release of radioactive concentrations in effluents in excess of limits, incidents involving an attempt to commit a theft or unlawful diversion of SNM, or to commit an act of sabotage of certain facilities, failure of safety systems, emergency core cooling or other related safety systems to perform their design function, or the MUF of SNM in excess of applicable limits, or similar matters.

(a) Orders to cease and desist

An order to cease and desist is ordinarily issued when a person is conducting unauthorized activities and has been notified of the need for authorization but fails to terminate the activity and other similar circumstances as appropriate.

(b) Orders to suspend a license

An order is ordinarily issued for immediate suspension of a license, or a portion thereof, as necessary to remove an immediate threat to the health, safety or interest of licensee's employees or the public, or to the common defense and security; or for noncompliance with AEC requirements relating to construction of a facility which, if not corrected immediately, could subsequently result in a significant threat to the health, safety or interest of employees or the public, or the common defense and security.

(c) Order to modify a license

An order for the modification of a license, in whole or in part, is ordinarily issued as an enforcement sanction when it is determined that a licensee's operations or activities must be limited or modified to protect the health, safety, or interest of the licensee's employees or the public, or the common defense and security.

(d) Orders to revoke a license

An order is ordinarily issued to revoke a license when:

1. The licensee's performance shows that he is not qualified to perform the activities covered by the license; or
2. Civil penalty proves to be ineffective as an enforcement action; or
3. The licensee refuses to correct items of noncompliance; or
4. A licensee does not respond to a "notice of violation"; or
5. A licensee's response to a "notice of violation" indicates inability or unwillingness to maintain compliance with regulatory requirements; or
6. Any material false statement is made in the application or in any statement of fact required under Section 182 of the Act.

(e) Denial of application for license renewal

Denial of an application for a license renewal is ordinarily used in lieu of an order for revocation where license renewal is pending or the expiration of the license term is imminent.

(f) Orders for other items of noncompliance

Orders to cease and desist, or for suspension, modification or revocation of a license are ordinarily issued for other comparable types of violations, infractions or deficiencies when the Commission deems such sanctions to be appropriate and necessary.

In all cases where orders are issued to impose civil penalties, to require a licensee to "cease and desist," or to suspend, modify, or revoke a license, the person so ordered may demand a hearing under 10 CFR Part 2. The hearing will be granted prior to implementation of the order except in cases where the Commission finds that the violation is deliberate or the public health, safety, or interest requires that the proposed action be temporarily effective pending the outcome of the hearing and/or further order.

REGULATORY OPERATIONS BULLETINS—CRITERIA

A Regulatory Operations Bulletin may be issued to a class of licensees requesting specific actions as a result of safety related equipment design inadequacies, defects, operating inadequacies, malfunctions, or failures of a generic nature that have occurred at a similar facility or operation. The Bulletin will specify that licensees inspect for and/or correct the inadequacies described in the Bulletin, notify Regulatory Operations of the corrective action taken or planned, and the date when action was or will be completed. An order may be issued if the response to a Bulletin is not prompt and effective.

IMMEDIATE ACTION LETTERS—CRITERIA

A Regulatory Operations Immediate Action Letter is ordinarily issued to solicit or confirm a licensee's commitment to certain actions for investigating, reporting, controlling, and correcting situations involvng defects, deviations,

failures, or administrative controls, at the licensee's facility. An order may be issued if the response to an Immediate Action Letter is not prompt and effective.

Categories of Items of Noncompliance

The Commission and representatives of the nuclear industry have recognized that the significance of items of noncompliance with AEC requirements varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment. The Commission considers that it is desirable to include in Notices of Violation an indication of the significance of each item of noncompliance cited. As a means of categorizing the items of noncompliance into an order of importance which will express their relative significance, the Commission has established three categories of items of noncompliance as follows:

Violation

A violation is an item of noncompliance of the type listed below, or an item of noncompliance (1) which has caused, contributed to or aggravated an incident of the type listed below, or (2) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., a situation where the preventive capability or controls were removed or otherwise not employed and created a substantial potential for an incident or occurrence with actual or potential consequences of the type listed below:

(*a*) Exposure of an individual in excess of the radiation dose specified in 10 CFR 20.403(b) or exposure of a group of individuals resulting in each individual receiving a radiation dose which exceeds the limits of 10 CFR 20.101 and a total dose for the group exceeding 25 man-rems.

(*b*) Radiation levels in unrestricted areas which exceed 50 times the regulatory limits.

(*c*) Release of radioactive materials in amounts which exceed specified limits, or concentrations of radioactive materials in effluents which exceed 50 times the regulatory limits.

(*d*) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is lost.

(*e*) Failure to function when required to perform the safety function or loss of integrity of a Seismic Category I system, or structure; or other component, system, or structure with a safety or consequences limiting function.

(*f*) Exceeding a safety limit as defined in technical specifications associated with facility licenses.

(*g*) Industrial sabotage of utilization or fuel facilities.

(*h*) Radiation or contamination levels in excess of limits on packages or loss of confinement of radioactive materials in packages offered for shipment on a common carrier.

(*i*) Diversion or theft of plutonium, uranium 233, or uranimum enriched in the isotope U-235.

(*j*) A breakdown in management or procedural controls as evidenced by items of noncompliance in several areas of the QA criteria and license requirements.

(*k*) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes a violation of the same importance level.

Infractions

An infraction is an item of noncompliance of the type listed below, or an item of noncompliance (1) which resulted in a reduction of preventive capability below requirements but redundant controls precluded an item of noncompliance of the violation category, or (2) which caused, contributed to or aggravated an incident of the type listed below, or (3) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., the preventive capability or controls were removed or otherwise not employed and there was substantial potential for an accident or occurrence with actual or potential consequences of the type listed below:

(*a*) Exposure of an individual or groups of individuals to radiation in excess of permissible limits but less than the values in 10 CFR 20.403.

(*b*) Release of radioactive materials in concentrations or rates which exceed permissible limits but in amounts less than permissible limits.

(c) Failure to function or loss of integrity of a Seismic Category I system or structure, or other component, system or structure with safety or consequences limiting function during test; or failure to meet surveillance frequencies.

(d) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is impaired.

(e) Exceeding limiting conditions for operation (LCO).

(f) Inadequate management or procedural controls.

(g) Safety system settings less conservative than limiting safety system settings.

(h) A quantity of SNM unaccounted for which exceeds permissible limits.

(i) Exceeding limits or limiting conditions for operation in licenses, technical specifications, guides, codes, or standards which are imposed for the purpose of minimizing adverse environmental impact.

(j) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes an item of noncompliance of the same category.

Deficiency

A deficiency is an item of noncompliance in which the threat to the health, safety, or interest of the public or the common defense and security is remote; and no undue expenditure of time or resources to implement corrective action is required; and deficiencies include such items as noncompliance with records, posting, or labeling requirements which are not serious enough to amount to infractions.

Failure to report deficiencies as required constitutes an item of noncompliance of the same category.

4. Apparently, the installation of the cable tray penetration sealant and fire retardant coating had not been completed prior to operations of Units 1 and 2 (Investigation Report—I-2). How did this manage to slip past detection by both TVA and NRC?

Response: Inspections conducted by NRC personnel using the inspection manual in effect prior to the Browns Ferry fire did not include checking of the cable penetrations or fire stops. The inspection manual has since been revised to require such inspections. Further, it is understood that TVA did not include such inspections within its quality assurance program.

5. The procedures for notifying States and local support agencies left something to be desired. How closely had NRC reviewed these procedures?

Response: The TVA Radiological Emergency Plan (REP) states that the responsibility for the coordination of a nuclear emergency with the appropriate public agencies is vested in the Central Emergency Control Center (CECC) Director.[1] During a nuclear emergency at a plant site the CECC staff will function to provide assistance as necessary to the site and division emergency organizations and will provide all information requested by outside agencies. The chain of notification is shown in Figure 1.[2]

When a nuclear emergency is detected at a site, the Site Emergency Director notifies the CECC Director who then notifies the appropriate federal, state, and local authorities.[3] The detailed procedures for this notification are found in Section III of the TVA-REP. These procedures list several conditions of gas release, liquid release, and abnormal radiation as action levels at which protective measures would be initiated.[4] As no action levels associated with these conditions were reached during the fire,[5,6] the CECC was not initially activated. Some increase in airborne activities did occur[7] as a result of the shutdown of the ventilation system but this situation did not constitute a nuclear emergency, as defined in the REP (radiological hazard to onsite personnel or radiation levels in excess of 10 CFR Part 20 values to the public). The REP was activated however, primarily to expedite communications and the decision making process.

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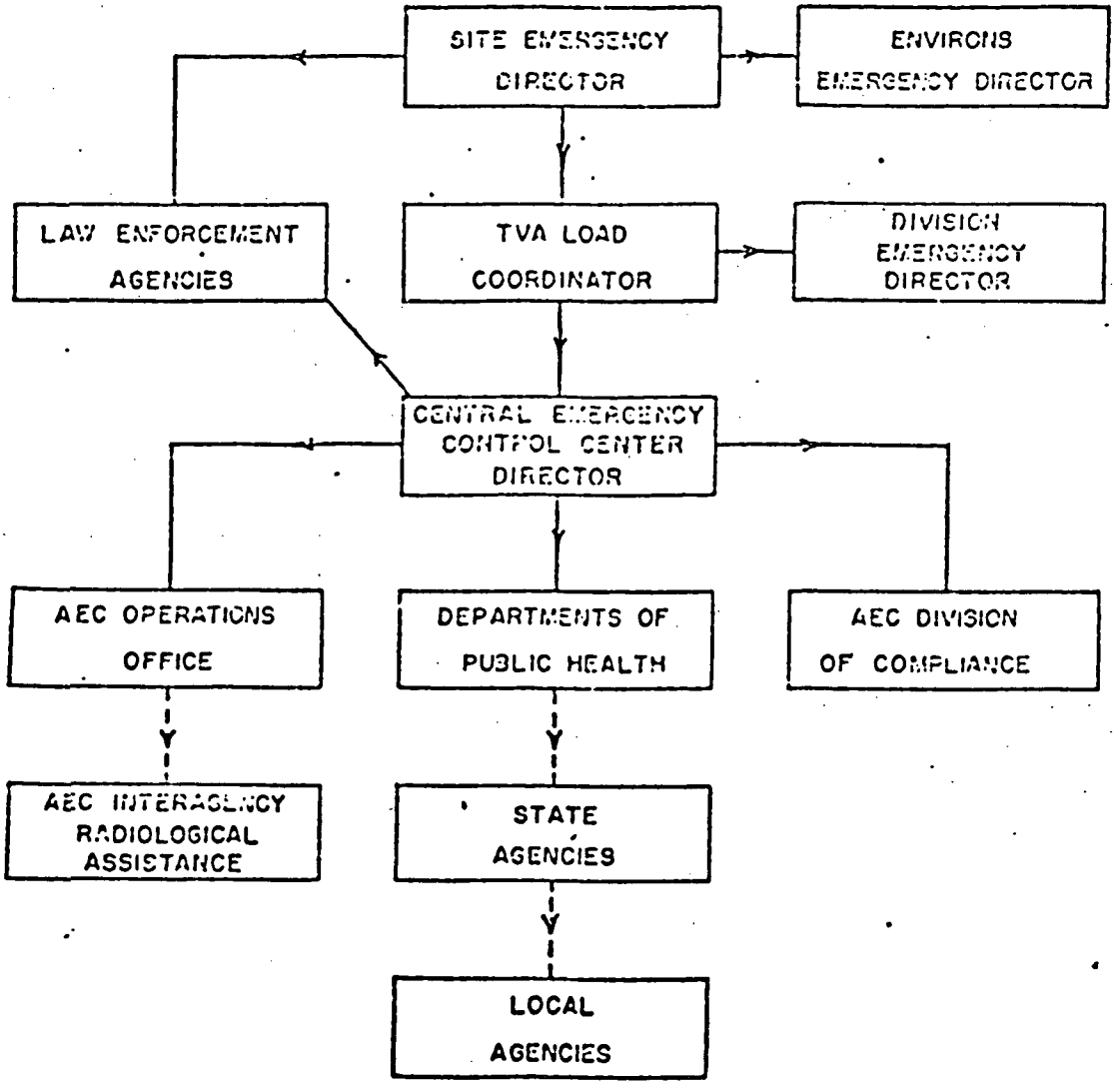
[1] TVA Radiological Emergency Plan for the Browns Ferry Nuclear Plant, revised June 5, 1972, page 37.
[2] Ibid, page 19.
[3] Ibid, page 7.
[4] Ibid, page 11.
[5] IE Report No: 50-259/75-1D (preliminary), page 4.
[6] IE Report No: 50-259/75-1B (preliminary), page 3.
[7] TVA Plan for Evaluation, Repair & Return to Service of Browns Ferry Unit 1 and 2, Section 111, page 43.

The NRC staff reviewed these procedures. It was found that:

1. Authorities and responsibilities of key individuals had been delineated;

Figure 1

CHAIN OF NOTIFICATION



——— TVA CONTACTS

- - - - INTRA-AGENCY CONTACTS

2. Communication links in the form of call lists had been established for notifying emergency personnel;

3. Action levels had been chosen for notification of offsite agencies in radiological emergency situations; and,

4. Written agreements with offsite emergency organizations had been provided.

We found that the emergency planning program for the Browns Ferry Nuclear Plant met the requirements of 10 CFR 50, Appendix E, and was acceptable. This finding was presented in the Safety Evaluation Report of June 26, 1972. In particular, the TVA-REP met the requirements of Part IV D of Appendix E, in that procedures for the notification of local, State, and Federal agencies were included in the plan and were found satisfactory.

Office of Inspection and Enforcement inspectors do not review procedures prepared by State and local support agencies. They do, however, verify statements made by the licensee in his emergency plan concerning his notification arrangements with State and local authorities by contacting the agencies identified. In the case of Browns Ferry, such contacts by inspectors were made on September 30, October 1–4, October 20–21, and November 1, 1974. These visits are reported in publicly available Inspection Reports.

The fact that the specified action levels contained in the TVA-REP were not reached may have been a factor in the manner in which the notification procedures were implemented. Apparent deficiencies in their implementation were reported to the licensee in the IE Inspection Report 50–259/75–1 dated July 25, 1975. The TVA responded to the alleged infractions and areas of concern by letter dated September 2, 1975.[8] These responses were discussed in IE letter dated September 29, 1975.[9]

6. How does NRC reconcile the need for water deluge systems in cable rooms (page 7 of Dr. Hanauer's testimony) with TVA's contention that the use of water might have led to further loss of control of the reactors?

Response: There is considerable room for judgment and professional disagreement on this issue. Mr. Green of TVA pointed out at the hearing the potential for water on the cables disabling electrical circuits that might otherwise operate successfully in spite of the fire damage. He stated his preference for waiting "until the situation had stabilized" and until equipment available for service had been evaluated before pouring water on these cables.

Past experience in other situations has indicated that it is generally preferable on a risk-benefit basis, to extinguish the fire as quickly as possible with whatever agents are needed. In this way the spread of damage can be more quickly and more surely limited. Delay in the use of water would necessarily incur the risk of additional equipment becoming unavilable due to further spread of the fire.

This is not to say that the indiscriminate use of water is required under all circumstances for all fires. Or that it is totally clear that the early use of water will result in less risks to the public for all fires involving nuclear power plants. Such decisions during an emergency must be left up to qualified personnel on the scene since operating manuals and procedures cannot detail all possible situations and events. However, water has often been needed to put out most electrical cable fires of more than trivial size. This view is also held by the Fire Underwriters and other fire experts we have consulted. They have informed us that current practice is to provide water deluge systems even for such complex systems as large computer installations. They state that this recommendation is based on a balance of the risks and benefits in fighting fires in electrical equipment and has been shown by experience to be preferable.

The course of the Browns Ferry fire (as shown in Figure 4 of Mr. Gilleland's testimony) shows a progressive onset of unavailability of various equipment. We believe that the spread of the fire made some contribution to this. For example, some of the redundant circuits disabled unexpectedly were in metal conduits that were overheated by the fire when it progressed to that location. It therefore appears to us that, in general, fires should be fought promptly and as effectively as possible including the use of water when necessary. Also, in hindsight, it is known that the electric shock considerations were not significant at Browns Ferry. The special hose nozzles available, when used, put out a divided spray that does not endanger the fireman holding the hose.

It is noteworthy that TVA now proposes to install water deluge systems at the Browns Ferry plants in regions of high concentration of electrical cables.

7. The Committee understands that a petition has been filed to change the NRC regulations to require plant operators to distribute emergency plans to the public and conduct yearly drills. What is the status of NRC's consideration of this matter?

Response: By letter dated August 6, 1975, the Public Interest Research Group and thirty other specified citizen groups petitioned the NRC to require licensees and applicants to instruct citizens in public evacuation procedures in case of major nuclear incidents and to test public evacuation plans in realistic drills. The petition has the docket number PRM–50–14 and was published in the Federal

[8] Letter TVA to NRC Region II of 9/2/75, pages 14, 25, 38, 39, and 40.
[9] Letter NRC Region II to TVA of 9/29/75.

Register on September 23, 1975, with a 60-day comment period. The NRC will consider public comments in its subsequent response to the petition.

8. Over the years, the primary emphasis in nuclear safety has been placed on hypothetical nuclear accidents, such as loss of coolant accidents and reactor excursions. The fire at Browns Ferry and other fires over the years would appear to suggest that the accidents that have occurred at nuclear plants have been more conventional in nature. In this regard,

(a) What consideration does NRC give to fires and other conventional accidents in its safety reviews?

(b) How is NRC organized to assure that such hazards are recognized: i.e., does NRC have any special expertise in fire detection and prevention?

(c) Does NRC intend to make any organizational changes to put increased emphasis on preventing fires and other conventional hazards?

Response (a)—pertaining to fires: The NRC review and evaluation of the fire protection of a proposed nuclear power plant includes the fire potential as described in the applicant's Safety Analysis Report (SAR), and a review of the design layout of the fire protection system (FPS) showing the characteristics and locations which define the "fire prevention" and "fire protection" portions of the system. The review also includes the description, identification and types of fire hazards that can exist and fire risk evaluations for each of the postulated hazards. Examples of subjects reviewed include:

1. the selection of fire fighting methods, manual or automatic equipment, and safety devices, including the detection, suppression, control, and extinguishing systems as described in the SAR;

2. the selection of appraisal and trend evaluation systems to be used, and the design of the fire detection and alarm system; and.

3. the general plans for performing inspection checks and the frequency of testing to maintain a reliable detection and alarm system.

The staff reviews the building and facility arrangements and structural design features which control selection of the methods for fire prevention, control and extinguishment, and control of fire hazards. Fire barriers, egress routes, fire walls, and the isolation and containment features are included in the review.

The staff determines from the SAR it appreciable amounts of combustibles are to be located on site, and reviews analyses of the effects of these hazards on safety-related equipment located nearby. The staff varifies that these analyses include a conservative selection of design basis fires, as determined from the types and quantities of stored combustible materials.

The functional performance of extinguishing systems, as described in the SAR, is reviewed to verify the adequacy of the FPS to protect electrical equipment. On multiple unit applications, the additional fire protection and control provisions during construction of the remaining units are reviewed to verify that the integrity of the fire protection system for the safe operation and shutdown of the facility is maintained.

Information that NRC has requested to be incorporated in the applicant's SAR is set forth in Regulatory Guide 1.70.4, "Additional Information Fire Protection Considerations for Nuclear Power Plants", February 1974.

Response (a)—other conventional accidents: The NRC, the vendors, and plant operators have always recognized that nuclear plants are subject to the same types of accidents that can affect any industrial facility. However, nuclear plants are different in that these conventional industrial accidents, if uncontrolled, could in some cases affect the cooling of the core and potentially result in the release of radioactive material.

The staff includes in its safety review an evaluation of these conventional types of accidents. A list of these accidents and the Standard Review Plans which detail the review of each, is attached. These other events include natural phenomena such as tornadoes, wind storms, floods, tsunamis, seiches, and earthquakes. The effect of accidents at nearby locations such as waterways, railways and roads that might produce explosions, missiles, smoke, or toxic or chemical containing gases are considered. At locations where the potential is significant, the consequences of aircraft crashes are considered. Accidents within the plant that might result in these effects are also considered. Radioactive spills are considered and, although not conventional, the causes of such events are usually conventional.

Additionally, a large portion of the safety review consists of evaluating the response of the plant to those transients to which every power plant, conventional or nuclear, is potentially subjected. Some of these transients are well within the normal operating conditions of the plant. However, more attention is directed toward abnormal anticipated transients, since their effects are more severe. Abnormal operational transients are those events which can be reasonably expected to occur during the life of the plant. These events include turbine trips, losses of feedwater, inadvertent actuation of equipment such as the ECCS, and other events which would result from operator errors or equipment malfunctions.

Response (b): Within the NRC staff, especially in the Offices of Nuclear Reactor Regulation and Inspection and Enforcement, personnel are assigned the responsibility for reviewing reactor applications and for inspecting reactor facilities to assure that various potential hazards are recognized, reviewed and evaluated. The purpose of these reviews and inspections is to determine whether plant design criteria and bases are adequate to assure safe reactor shutdown in the event of the occurrence of any of the postulated hazards. Some of these "conventional hazards" considered are fires, both inside and outside the plant, natural phenomena, aircraft crash and its fire effects (where applicable), and/or water surface transportation accidents that could involve fire or explosions and hazards that may be related to conditions surrounding the reactor site.

The NRC does have expertise in the areas of fire detection and prevention. However, if an unusual situation is noted by the staff, consultants are available and used from other federal agencies to assist in the resolution of problems. Agencies that have been or are being used for this service include ERDA, NASA and the Coast Guard.

Response (c): No organizational changes are planned to place increased emphasis on preventing fires and other conventional hazards. However, increased attention is being given to fire hazards in already established organizational components that review fire prevention and protection and other conventional hazards. The Office of Inspection and Enforcement has developed guidance for inspectors in the area of fire protection equipment and procedures. NRC has under consideration the degree of recognition which can be given to inspections performed by the insurance groups as third party inspections.

Task forces to review and evaluate the Browns Ferry fire have been established and have been involved since the fire occurred. The work being done by these groups is being fed back into the existing NRC organization. The staff involved in evaluating the Browns Ferry fire has been augmented by consultants. These consultants are assisting in the evaluation of the fire, in the development of criteria and guides for use in evaluation of fire protection programs for nuclear power plants, and are conducting tests on various aspects of fire conditions that require more detailed knowledge, and providing general advice on fire protection.

Listing of Postulated Accidents

Accident and standard review plan	*Branch with primary responsibility*
Accidents at nearby locations (smoke, chemicals, gases): 2.2.3—Evaluation of potential accidents.	Accident Analysis.
Storms:	
2.3—Meteorology	Site analysis.
3.3—Wind and tornado loadings	Structural engineering.
Earthquakes:	
2.5—Geology and seismology	Site analysis.
3.2—Classification of structures, components and systems.	Reactor systems.
3.7—Seismic design of structures	Structural engineering.
3.8—Design of category I structures	Do.
3.10—Seismic design of category I instrumentation and electrical equipment.	Electrical, instrumentation and control systems.
Floods:	
2.4—Hydraulic engineering	Site analysis.
3.4—Water level (flood) design	Auxiliary and lower conversion systems.

Listing of Postulated Accidents—Continued

Accident and standard review plan	*Branch with primary responsibility*
Missiles:	
3.5.1.1—Internally generated missiles (outside containment).	Auxiliary and lower conversion systems.
3.5.3—Barrier design procedures	Structural engineering.
3.5.1.2—Internally generated missiles (inside containment).	Reactor systems.
3.5.1.4—Missiles generated by natural phenomena.	Accident analysis.
3.5.1.5—Site proximity missiles (except aircraft).	Do.
3.5.1.6—Aircraft hazards	Do.
3.5.2—Structures, systems and components to be protected from externally generated missiles.	Auxiliary and power conversion systems.
Anticipated transients:	
15.1.1-15.1A—Decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a steam generator relief or safety value.	Reactor systems.
15.1.5—Spectrum of steam system piping failures inside and outside of containment (PWR).	Do.
15.2.1-15.2.5—Steam pressure regulator failure (closed), loss of external load or turbine trip, closure of main steam isolation valve (BWR) and loss of condenser vacuum.	Do.
15.2.6—Loss of nonemergency A-C power to the station auxiliaries.	Do.
15.2.7—Loss of normal feedwater flow.	Do.
15.2.8—Feedwater system pipe breaks inside and outside containment (PWR).	Do.
15.3.1-15.3.2—Loss of forced reactor coolant flow including trip of pump and flow controller malfunctions.	Do.
15.3.3-15.3.4—Reactor coolant pump seizure and reactor coolant pump shaft break.	Do.
15.4.4-15.4.5—Startup of inactive loop or recirculation loop at an incorrect temperature, and flow controller malfunction causing an increase in BWR core flow rate.	Do.
15.4.6—CVCS malfunction that results in a decrease in boron concentration in the reactor coolant (PWR).	Do.
15.6.1—Inadvertent opening of a PWR pressurizer safety/relief valve or a BWR safety/relief valve.	Do.
15.7.3—Postulated radioactive releases due to liquid-containing tank failures.	Effluent treatment systems.
15.5.1-15.5.2—Inadvertent operation of ECCS and CNCS malfunction that increases reactor coolant inventory.	Reactor systems.
15.6.2—Failure of small lines carrying primary coolant outside containment.	Accident analysis.

Listing of Postulated Accidents—Continued

Accident and standard review plan	*Branch with primary responsibility*
Anticipated transients—Continued	
15.6.3—Radiological consequences of steam generator tube failure (PWR).	Do.
15.7.1—Waste gas system failure	Do.
15.7.5—Spent fuel cask drop accidents.	Do.
15.7.2—Radioactive liquid waste system leak or failure.	Do.
15.4.1—Uncontrolled control rod withdrawal from a subcritical or low power startup condition.	Reactor systems.
15.4.2—Uncontrolled control rod assembly withdrawal at power.	Do.
15.4.3—Control rod misoperation (system malfunction or operator error).	Core performance.
15.4.7—Inadvertent load and operation of a fuel assembly in an improper position.	Do.
15.4.8—Spectrum of rod ejection accidents (PWR).	Do.
15.4.9—Spectrum of rod drop accidents (BWR).	Do.
15.6.4—Radiological consequences of main steam line failure outside containment (BWR).	Accident analysis.
15.6.5—Loss-of-coolant accidents resulting from spectrum of postulated piping breaks within the reactor coolant pressure boundary.	Reactor systems.
15.8—Anticipated transients without scram.	Do.

9. Mr. David Comey has made two specific recommendations to NRC as a result of the fire:

(*a*) That operation of nuclear units not be permitted at stations where additional plants are under construction.

(*b*) That all reactors not fully complying with IEEE standards on separation of electrical cables be shut down.

What is the status of NRC's actions on these recommendations?*

Response (a): Mr. David Comey's recommendation "that licensees be required to shut down operating units of nuclear power units at multi-unit facilities during periods when work on a unit under construction could compromise the integrity of the engineered safety features of an operating unit or units", has been treated as a Petition for Rulemaking since it involves a potential change to NRC Regulations. The Petition was published in the Federal Register on May 9, 1975, with comments requested from the public by July 8, 1975. Three comment letters were received. We expect to respond to this petition, which will include consideration of these comments, by February 1, 1976.

Response (b): On March 24, 1975, Mr. David Comey, citing the March 22, 1975 occurrences at Browns Ferry, asked the NRC to identify those operating plants that do not fully comply with IEEE–279 and to show cause why those plants should not be shut down until they are retrofitted to meet that standard.

On May 5, 1975, NRC provided Mr. Comey with the identity of these eight plants which, since they were issued operating licenses prior to the issuance of IEEE–279, were evaluated against pre-IEEE–279 criteria. In that letter, he was informed of steps that had been and would be taken that in our judgment provide reasonable assurance that all operating plants can continue to operate without undue risk to the health and safety of the public pending completion of the ongoing NRC investigation of the Browns Ferry fire.

In response to this information provided by the NRC, on June 16, 1975, Mr. Comey limited his March 24, 1975 request to a review by the NRC staff of two

*See Appendix 13 for exchange of correspondence between NRC and David Comey.

plants, Indian Point 1 and Dresden 1, pending completion of the report by the NRC Special Review Group on the investigation of the fire.

On August 4, 1975, the NRC staff informed Mr. Comey that on June 11, 1973 the Consolidated Edison Company had been directed pursuant to 10 CFR 50.109 to make certain design changes on Indian Point 1 based on a comparison of the as-built plant protection system with the criteria of IEEE-279. Indian Point 1 has been shut down since October 31, 1974.

Our August 4, 1974 letter also stated that on July 31, 1975 Commonwealth Edison had been requested to compare the as-built design of that plant with the IEEE-279 criteria (1968 Edition); to identify features that do not conform to that standard; and to determine whether, based on its evaluation, there are changes in the facility which could be accomplished that provide substantial additional protection which is required for public health and safety. Commonwealth Edison provided its report on the matter on October 1, 1975. The staff is now evaluating the report to determine what actions, if any, are necessary on that plant to protect the health and safety of the public.

10. What is NRC's position regarding the continuing applicability of Technical Specifications in the event some departure from them is required as the result of an on-site emergency?

Response: Technical Specifications for each facility are required by Commission Regulations (Paragraph 50.36). These specifications set forth limitations and restrictions needed to maintain a facility in a safe condition during plant operation. They further set forth requirements on equipment operability for the purpose of assuring that the requisite number of components are operable in the event of postulated accidents.

Technical Specifications are not expected to cover all conceivable conditions that might arise during an on-site emergency. Should an emergency condition arise which, in the interest of public health and safety, required action inconsistent with a Technical Specification, we would not construe the Specification to be applicable.

11. Please provide background on why the polyurethane material of the type tested by the applicant was not the type used. Also, discuss why these tests were relied upon, when it has been documented that small scale laboratory tests are inadequate to determine the fire properties of a material in large-scale applications. (AEC Serious Accidents Bulletin No. 336, September 19, 1974). Finally, why was polyurethane foam approved for use in such an inaccessible location, when the fire hazards of this practice have long been known? (AEC Health and Safety Bulletin No. 170, dated September 10, 1963 recommends that consideration be given to attaching polyurethane so that it can be removed more easily in case of fire.)

Response: The pieces of sheet polyurethane used to seal the leaks and provide a dam prior to placement of the polyurethane foam represented material not tested by TVA. Approval to use the untested sheet polyurethane in the penetration sealing activity was given by one TVA employee (NRC Investigation Report, p I-10). This individual was aware that the sheet polyurethane had not been tested (NRC Investigation Report, Statement EE-FF). He also was aware of the TVA flammability tests of the sealant material (another form of polyurethane) covered with the flame-resistant material.

In reply to the NRC Notice of Violation concerning the alleged failure of TVA to have detailed written procedures approved by the plant superintendent and reviewed by the TVA Plant Operation Review committee for this penetration sealing activity, TVA replied ". . . it was not recognized that this work on the penetration was of such a nature as to have an effect on the safety of the reactor, on surveillance or testing requirements, or had the potential for release or radioactivity; therefore, leak testing sealing, and inspection of the penetrations were not accomplished by detailed, approved, written procedures as would be required for safety-related items." (p 3, of Enclosed Notice of Violation, to TVA letter of September 2, 1975, from J. E. Gilleland, TVA, to Norman C. Moseley, NRC).

TVA conducted tests of cable penetration fire stops in May 1973 for the purpose of examining the effectiveness of urethane pressure seals. A copy of the TVA evaluation memo concerning these tests was provided as Exhibit A4 in the I&E report of July 25, 1975. The staff does not consider these tests to be representative of the actual installation in that a differential pressure was not utilized to simulate actual in-plant conditions. TVA is now conducting additional tests to more closely simulate the as-built penetrations; application of pressure will be included. The purpose of these tests is to determine whether properly con-

structed fire stops of the original design are effective and whether those fire stops not required to be replaced due to restoration work can be left in place.

The inaccessibility of the location was, indeed, a handicap in performing any work or testing on the seals at this location. This inaccessibility contributed to the ignition of the fire and to the difficulty in extinguishing it immediately (as had been done for other similar small fires). However, it should be noted that a properly constructed seal was shown to have been an effective fire stop and it should have been possible, and will be accomplished in the future, to devise procedures for working on and testing these seals even in relatively inaccessible locations without any significant fire hazard.

APPENDIX 10

QUESTIONS AND ANSWERS: JCAE/TVA

LETTER DATED SEPTEMBER 17, 1975, MURPHY TO WAGNER, WITH ADDITIONAL QUESTIONS TO TVA

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 17, 1975.

Mr. AUBREY J. WAGNER,
Chairman, Board of Directors,
Tennessee Valley Authority, Knoxville, Tenn.

DEAR MR. WAGNER: Enclosed is the transcript of the September 16, 1975 hearing on the Browns Ferry fire. The TVA witnesses should review and make any necessary corrections of your testimony in accordance with the instructions attached to the transcript. Also enclosed is a list of additional questions on the TVA testimony which the Committee would like you to provide answers to for the record.

Since we are endeavoring to publish and distribute the hearing as rapidly as possible, it would be appreciated if you would return the corrected transcript and your answers to the additional questions by no later than September 29, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, Jr.,
Executive Director.

QUESTIONS FOR TVA WITNESSES

Browns Ferry Hearing—September 16, 1975

1. How close, in TVA's view, was the plant to having a serious accident—such as fuel damage or core meltdown—which might have presented a serious public health and safety problem? Describe briefly and factually the equipment remaining in operation or which could be made operable within the necessary time frame to prevent core damage. Summarize both the regular and "emergency" function of this equipment. Show whether the availability of and need for this equipment was perceived during the time of the accident, and state the degree of confidence in whether this equipment would actually have been used, if needed.

2. There had been a past history of small fires at the Browns Ferry plant. Why hadn't preventive actions been taken by TVA? (Inspection Report, page I-11)

3. Apparently, there was considerable difficulty with finding sufficient operable air breathing equipment (Inspection Report I-13). Why hadn't TVA fire inspectors detected this situation in their normal periodic inspections of fire equipment?

4. Does TVA have an internal safety review committee to assess plant operating practices? At what level of management was the open flame inspection procedure approved?

5. How often have NRC inspection and enforcement personnel visited the plant and when was the last such visit? What infractions were reported at that time, and were any of these still uncorrected at the time of the fire? If so, did any of these contribute to the accident or its aftermath in any way?

6. Why had the material around the penetration in question not been coated with flamemastic? Why had this penetration not been adequately filled with foam in the first place? Was operation of the plant under these conditions in violation of NRC regulations?

7. Do you consider that TVA was adequately prepared to fight this fire? Are you making any changes to fire control plans and procedures as a result of this fire? Are there any actions that should have been taken by NRC or the State that might have enabled TVA to better respond to the situation?

8. What improvements does TVA believe desirable to make in the procedures used to notify the State of Alabama health officials and other local and county civil defense and police officials in the event of incidents like the Browns Ferry fire?

9. According to the NRC report (II-7), there was considerable concern on the part of the operators that the water level might drop below the top of active fuel. Is that an accurate statement, and what would the consequences have been if the fuel had indeed been uncovered?

10. You note on page 4 that both Units 1 and 2 were shut down shortly after anomalous events began to occur. Could this fire, or any fire, cause an inability to scram? What is the basis for your response?

11. What has TVA's experience been with respect to fires in its fossil fired units? How does this compare with that of its nuclear units?

[Subsequent TVA response follows:]

TENNESSEE VALLEY AUTHORITY
KNOXVILLE, TENNESSEE 37902

September 24, 1975

The Honorable George F. Murphy, Jr.
Executive Director
Joint Committee on Atomic Energy
Washington, D.C. 20510

Dear Mr. Murphy:

In accordance with your request of September 17, 1975, we have reviewed the transcript of the September 16, 1975, Joint Committee hearing on the Browns Ferry fire and prepared answers to your additional questions. Our comments and responses are enclosed.

Very truly yours,

Aubrey J. Wagner
Chairman of the Board

Enclosure

Enclosure

Question 1

How close, in TVA's view, was the plant to having a serious accident--such as fuel damage or core meltdown--which might have presented a serious public health and safety problem? Describe briefly and factually the equipment remaining in operation or which could be made operable within the necessary time frame to prevent core damage. Summarize both the regular and "emergency" function of this equipment. Show whether the availability of and need for this equipment was perceived during the time of the accident, and state the degree of confidence in whether this equipment would actually have been used, if needed.

Answer

Because there was always more than one means available to supply water to the reactor in quantities sufficient to keep the core covered, the likelihood of fuel damage due to the fire event was, for all practical purposes, nonexistent. With the ability to keep the reactor fuel covered with water and with primary and secondary containment intact, there was no serious threat to public health and safety. The following is a summary of equipment remaining in operation or which could be made available during the fire event. These evaluations are restricted to unit 1 since a review of the events makes it clear that the adverse effects of the fire on unit 2 were far less than on unit 1 and at all stages of the incident unit 2 had more capability for maintaining a safe shutdown condition. In addition, due to a longer operating history, the decay heat removal requirements were greater for unit 1. The capability of many of the following components depends on the reactor system pressure. For reference pressures see figure 5 of the statement of Jack E. Gilleland.

Steam turbine-driven feedwater pumps - 3 pumps - capacity 10,800 gpm each

The function of the feedwater pumps is to supply water to the reactor during power operation and during a period following reactor shutdown while steam pressure above 350 pounds still exists. The feedwater pumps have no emergency function. These pumps were available at the time the fire occurred and remained

Question 1 (continued)

available and in use until approximately five minutes after reactor shutdown when they became unavailable due to closure of the main steam isolation valves which cut off their steam supply. The status of the feedwater pumps was known to the operator at all times. TVA systems analyses undertaken subsequent to the fire event have revealed that there were alternate means available by which the main steam isolation valves could have been opened, thus making the steam-driven feedwater pumps available. This was not considered by the plant operator during the fire event.

Condensate Booster Pumps -- 3 pumps - capacity 10,800 gpm each

Three condensate booster pumps have the normal function of supplying water to the steam-driven feedwater pumps and supplying water directly to the reactor at pressures less than 435 pounds per square inch. These pumps have no assigned emergency function, but were available throughout the fire event and one of these pumps was in continuous operation and supplied water to the reactor after its pressure was reduced.

Condensate Hot-Well Pumps - 3 pumps - capacity 10,800 gpm each

The normal purpose of these pumps is to supply water to the condensate booster pumps during normal operation. They have the capability of supplying water directly to the reactor at pressures less than 160 pounds per square inch. They have no assigned emergency function. These pumps were available and two were in continuous operation throughout the fire event. The status of all three pumps was known to the operator.

Question 1 (continued)

Control Rod Drive Pumps - one per unit plus one shared spare - capacity 225 gpm each

Normally, one control rod drive pump is aligned to each reactor and one spare is available which may be aligned to either reactor. The purpose of these pumps is to supply operating water under pressure to the control rod drive units as well as to supply cooling water for the control rod drives. In normal operation, each of these pumps supplies 100 gpm of water in the reactor while performing its assigned function. It has no emergency function. During the fire event, one control rod drive pump was aligned to each reactor and ran continuously throughout the event. Using controls located in the control room, the operator increased the unit 1 pump flow output to approximately 130 gpm. Had the operator opened an accessible manual valve in the reactor building, the flow could have been increased to 225 gpm. The availability of the common spare pump was known but the pump was not operated. In addition, the unit 2 pump could have been aligned to unit 1, but the operator did not perceive this mode of operating at that time. However, as indicated in our testimony, the unit 1 pump alone would have been adequate.

Standby Liquid Control Pumps - 2 pumps - capacity 56 gpm each

Standby liquid control pumps had the assigned emergency function of injecting a boron solution into a reactor at up to full pressure should the normal shutdown method fail. Using a test mode, these pumps could have been used to supply in excess of 100 gpm of demineralized water on each reactor. These pumps were not used but the operator was aware of this capability.

Question 1 (continued)

Reactor Core Isolation Coolant - one pump - capacity 600 gpm

The RCIC system is not considered a part of the nuclear safety systems but provides high-pressure makeup water to the reactor vessel if the feedwater system does not operate properly or if the reactor becomes isolated and the coolant inventory is decreasing. The pump is steam driven, normally using steam from the reactor vessel but in a test mode may be operated using steam from an auxiliary boiler. This requires installation of a short piece of pipe available for that purpose which requires approximately one hour to install. The RCIC was initially available and was shut down by the operator since it was not required. Approximately five minutes after reactor shutdown, the RCIC system became unavailable in its normal configuration due to closure of a steam supply valve. It appears that this system could have been made available within an hour had the operator made the decision to install the spool piece to establish a steam flow path from the auxiliary boiler. During the fire event, the operator did place the auxiliary boiler into operation and bring it to full steam pressure.

High Pressure Coolant Injection - one pump - 5,000 gpm capacity

The high pressure coolant injection system operates using reactor steam as a driving force. It has no normal function but has the emergency function of supplying high pressure water to the reactor during a loss of coolant accident. The HPCI was initially available to the operator but was shut down since it was not required. The HPCI became unavailable approximately five minutes after reactor shutdown and was unavailable throughout the remainder of the fire event.

Question 1 (continued)

Residual Heat Removal - 4 pumps - 10,000 gpm capacity each

The RHR is a multipurpose system. In the low pressure coolant injection mode, it is intended to supply water to the reactor at pressure less than 300 pounds during the loss of coolant event. In the torus cooling mode, it has the normal and emergency purpose of circulating suppression chamber water through heat exchangers as a means of controlling suppression chamber temperatures. It has the capability of spraying water into the drywell during a loss of coolant accident thereby limiting containment pressure and temperature. In the shutdown cooling mode, with reactor vessel pressure below 100 psig, it circulates reactor vessel water through a heat exchanger and back to the pressure vessel. The RHR system was initially available in all modes. During the fire, damage to valve and pump wiring disabled two of the RHR pumps. Use of the other two pumps was temporarily lost until power was restored to the valves which were required to line up the system for operation. At approximately 4:30 p.m., two RHR pumps were aligned in the torus cocling mode but were not used because of the possibility that during the valve alignment without position indication the pump discharge piping may have been drained and to start a pump on an empty pipe had the potential of system damage. The status of the system was known to the operator but conditions never were such that use of the available components under these circumstances was justified. The RHR system was placed into service in both the torus cooling and shutdown cooling modes later in the night after a procedure was prepared to manually align and check the systems under existing conditions.

Question 1 (continued)

Standby Coolant Supply - 2 pumps - 4,500 gpm capacity each

Browns Ferry has the capability to supply river water directly to the reactor, at pressures less than 185 pounds using RHR service water pumps. These pumps ran continuously throughout the fire event and this capability was known to the operator. Circumstances did not justify use of this extreme measure since use of river water instead of clean dimeralized water might jeopardize future operability of the reactor. The piping arrangement of this system is such that had the RHRSW pumps become inoperable, any other pumping means such as a fire truck could have been used to pump any available water, including river water, into the reactor through this system. Although this capability was recognized, its use was not seriously considered.

Core Spray System - 4 pumps - 3,100 gpm capacity each

The core spray system is provided to supply coolant to the reactor at pressures less than 300 pounds during the loss of coolant accident event. The core spray system initially aligned and was shut down by the operator because it was not needed. Two of the core spray pumps became unavailable due to the fire damage and the position of the valves in the core spray system was not known to the operator because of loss of position indication. After the fire, it was determined that had the remaining two core spray pumps been operated, water injection into the reactor would have occurred. Core spray pump operation was not an avenue pursued by the operator during the fire event, but the operator did give consideration to trying it in the event other means to supply water to the reactor were lost.

Question 1 (continued)

Condensate Transfer Pumps - 2 pumps - 1,000 gpm each

One of the normal functions of the two condensate transfer pumps is to supply charging water to the discharge piping of the core spray and RHR pumps. The system could have been used to put water into the reactor vessel at low pressure, less than 85 psi, by closing a head tank isolation valve and opening the inboard isolation valve on either core spray or RHR system. These pumps were not used, but the operator was aware of this capability.

Question 2

There had been a past history of small fires at the Browns Ferry Plant. Why hadn't preventive actions been taken by TVA? (Inspection Report, page I-11)

Answer

The plant operators were aware of one small fire which had occurred on Thursday, March 20, 1975. This fire had been discussed in a meeting of plant supervisors on Friday, March 21, and investigative efforts were in progress. Plant management was not aware of any other minor fires which may have occurred in this material because they were not reported. TVA has since made procedural changes controlling the use of open flames, fire reporting, and use of fire attendants when activities with a fire potential are in progress.

Question 3

Apparently, there was considerable difficulty with finding sufficient operable air breathing equipment (Inspection Report I-13). Why hadn't TVA fire inspectors detected this situation in their normal periodic inspections of fire equipment?

Answer

TVA believes that the emergency breathing apparatus was well maintained. Plant procedures for the care of this equipment were in effect and were being used, and the specified quantity of equipment was available.

Question 3 (continued)

Some air cylinders which were in the shop for reconditioning and were specifically marked were inadvertently brought to the scene of the fire, but they were not used and did not endanger personnel safety.

TVA investigations identified some problems with the air masks consisting of service life limitations, recharging deficiencies, and isolated cases of improper operator use. TVA has taken action to resolve these difficulties.

TVA investigations did not identify any deficiencies associated with the plant's inspection and maintenance program for breathing apparatus.

Question 4

Does TVA have an internal safety review committee to assess plant operating practices? At what level of management was the open flame inspection procedure approved?

Answer

TVA does have internal committees to review operating practices. At the time of the fire, procedures required review of construction activities by a licensed senior reactor operator to determine if safety considerations were involved. The senior reactor operator was empowered to authorize activities not having safety significance. Those activities recognized as having safety significance were reviewed by the Plant Operations Review Committee to determine if the necessary prerequisites have been met to authorize their performance. The leak test activities in progress at the time of the fire initiation had been reviewed and approved by the senior reactor operator who did not recognize the safety significance of this activity

Question 4 (continued)

and did not recognize that an open flame would be used. TVA has made procedural changes requiring plant supervisory review of all significant activities and review by the plant quality assurance staff to verify that all significant activities are supported with adequate instructions necessary for their performance. These reviews are a prerequisite to activity authorization by the plant superintendent.

Question 5

How often have NRC inspection and enforcement personnel visited the plant and when was the last such visit? What infractions were reported at that time, and were any of these still uncorrected at the time of the fire? If so, did any of these contribute to the accident or its aftermath in any way?

Answer

The NRC inspection and enforcement personnel have inspected the operation and construction activities at Browns Ferry Nuclear Plant 206 times prior to March 22, 1975. Prior to the fire, the NRC inspectors averaged three inspections per month. The last visits by the NRC inspectors prior to the fire were on February 25-28 and March 21, 1975. The summary of findings of that inspection and TVA's responses to those findings are attached. The only unresolved infraction concerned the format and method of approval of certain procedures relating to the nonradiological monitoring program. None of these items were related in any manner to the fire event.

ATTACHMENT TO QUESTION 5 AND ANSWER

Excerpt from NRC Inspection Report from Visit of February 25-28 and March 20, 1975

SUMMARY OF FINDINGS

I. Enforcement Items

A. Infractions

1. Contrary to Appendix B Technical Specification 2.2.3, the total residual chlorine in the condenser cooling water discharge was not monitored on a weekly basis when the auxiliary raw cooling water systems were being chlorinated.

   This infraction had the potential for causing or contributing to an occurrence related to health and safety.

2. Contrary to Appendix B Technical Specification 5.5.1 written procedures for the in-plant nonradiological monitoring program had not been prepared by the Division of Power Production.

   This infraction had the potential for causing or contributing to an occurrence related to health and safety.

B. Deficiencies

1. Contrary to Appendix B Technical Specification 5.3.2 an audit of the in-plant nonradiological environmental monitoring program had not been conducted.

2. Contrary to Appendix A Technical Specification 6.3.A.1 the reactor was twice started up on a period shorter than the period permitted by the startup operating procedure.

TVA Responses to NRC Inspection Report from Visit of February 25-28 and March 21, 1975

1.A.1 - The technical specifications required that total residual chlorine in the condenser cooling water discharge be determined weekly during periods when the raw cooling water system was being chlorinated. We met this specification by measuring the concentration in the auxiliary raw cooling water flows and computing the concentration in the main condenser cooling water discharge to the river based upon the relative flow volumes of the two systems. Because of the very low residual chlorine concentration in the condenser cooling water discharge, direct measurement with accuracy was impractical using standard instrumentation and techniques. To avoid future misunderstandings with the Inspection and Enforcement Branch, the technical specifications have been revised to specifically authorize the method in use for determining residual chlorine levels. We deny that the alleged infraction had the potential for causing or contributing to an occurrence related to health and safety.

ATTACHMENT TO QUESTION 5 AND ANSWER

I.A.2 - Written procedures were in effect for the in-plant nonradiological monitoring program but had been approved by plant section supervisors rather than the plant superintendent. A system of environmental surveillance instructions is being assembled to ensure uniformity and to provide a means to document requirement compliance. These instructions will be approved by the plant superintendent and will be completed prior to the return of any unit to service.

I.B.1 - TVA was in error in not conducting this audit during calendar year 1974. This was due, in part, to the ever-changing audit requirements. At the time this deficiency was noted, the audit was scheduled to be conducted in the second quarter of calendar year 1975. The audit was conducted on schedule and will be conducted annually hereafter.

I.B.2 - Except for minor details, the facts contained in the NRC report concerning this alleged deficiency are essentially accurate. At the time of the occurrence, the referenced operating instruction read as follows: "Observe the period meter when pulling rods, and govern withdrawal rate to avoid having a period shorter than 60 seconds." The operator was following the procedure in that he was observing the period meter and power level indication when pulling rods and was governing rod withdrawal in the most conservative, prudent manner with a one-notch-at-a-time withdrawal, waiting between notches. The reactivity change resulting from the smallest incremental movement between notches 12 and 14 was sufficient to give a period shorter than 60 seconds. The operator immediately inserted the rod a sufficient amount to stop the power increase. Following confirmation of this high notch worth, the rod program was revised. At no time was rod worth great enough to be a significant safety concern. The operating procedure has been revised in a less conservative direction to permit shorter periods without threat of violation.

Question 6

Why had the material around the penetration in question not been coated with flamemastic? Why had this penetration not been adequately filled with them in the first place? Was operation of the plant under these conditions in violation of NRC regulations?

Answer

Flamemastic was not present on all penetrations because it had been necessary to run additional cable through some penetrations during the accomplishment of plant improvement modifications. It was work prerequisite to reapplication of flamemastic material that led to the fire ignition.

The primary purpose of the polyurethane material was to seal around conductors leading into the secondary containment area. There was no requirement that individual penetrations be leak tight, but there was a requirement that the licensee be able to maintain 1/4 inch of water vacuum in the secondary containment with an in-leakage of less than 9,000 cfm. This requirement was always met.

The design of the penetration seal provided for coating completed penetrations with a flamemastic material. The fire barrier function of the flamemastic coated penetration seals is not mentioned in the Final Safety Analysis Report or other license documents. Because of this, TVA did not consider operation of the plant, under the conditions which existed, to be in violation of NRC regulations.

Question 7

Do you consider that TVA was adequately prepared to fight this fire? Are you making any changes to fire control plans and procedures as a result of this fire? Are there any actions that should have been taken by NRC or the state that might have enabled TVA to better respond to the situation?

Question 7 (continued)

Answer

TVA exhibited adequate preparation for handling a fire emergency situation by effectively implementing emergency plans and actions that ensured reactor safety and control of the fire. Reactor safety was of paramount importance and was assured at all times. The fire in the cable spreading room was controlled and suppressed at all times until extinguishment occurred. The fire in the reactor building remained in an extremely slow propagating, controlled condition until extinguishment occurred. Although preparations were adequate and effectively implemented, the cable tray fire has resulted in a reappraisal of TVA's program of fire prevention and protection for Browns Ferry. Immediate actions taken to improve fire protection at Browns Ferry have included expanding and improving the fire training of fire brigade members and leaders; establishing controls for cutting, welding, and open flames; providing trained fire watches in special areas during restoration work; providing additional portable fire-fighting equipment; and improving the scope and frequency of fire surveys. Design studies have been conducted to identify needed improvements in the fixed fire protection systems, fire water systems and related fire protection such as sealing penetrations, fire walls, etc.

TVA knows of no action that could have been taken by NRC or the State that would have improved our response and handling of the fire emergency at Browns Ferry.

Question 8

What improvements does TVA believe desirable to make in the procedures used to notify the State of Alabama health officials and other local and county civil defense and police officials in the event of incidents like the Browns Ferry fire?

Question 8 (continued)

Answer

Notification of State of Alabama health officials and other local officials of the Browns Ferry fire was carried out in accordance with the provisions in the Radiological Emergency Plan. These notifications were effective, and TVA does not plan to make any changes to the notification procedures. On September 18, 1975, TVA staff met with representatives of all local, county, and state agencies required to respond to an emergency at Browns Ferry. The primary discussion involved prompt notification of appropriate groups. Additionally, periodic reorientation meetings (at least annually) will be held with these organizations.

Question 9

According to the NRC report (II-7), there was considerable concern on the part of the operators that the water level might drop below the top of active fuel. Is that an accurate statement, and what would the consequences have been if the fuel had indeed been uncovered?

Answer

Maintaining an adequate water level in the reactor at all times is a primary concern of the operators. However, as indicated in the TVA testimony, the reactor cores were adequately cooled throughout the event and the water level remained above the reactor fuel. The reactor fuel can be uncovered for short periods of time without exceeding fuel cladding temperature of 2200° F. At this temperature, the core geometry is still amenable to cooling and the cladding would not fail upon the subsequent quenching by the addition of cooling water. Studies also show that keeping the water level above 2/3 of the active fuel is sufficient to limit the peak temperature to below 900° F.

Question 10

You note on page 4 that both units 1 and 2 were shut down shortly after anomalous events began to occur. Could this fire, or any fire, cause an inability to scram? What is the basis for your response?

Answer

The March 22 fire did not result in the loss of the ability to scram nor will any other fire result in the loss of the ability to scram. This response is based on the diversity and separation designed into the reactor protection system. The reactor protection system is fail-safe in that the failure modes in any of the logic and control circuits that would be involved in a fire results in a scram. Loss of electric power to the reactor protection system results in a scram.

Question 11

What has TVA's experience been with respect to fires in its fossil-fired units? How does this compare with that of its nuclear units?

Answer

TVA's overall experience with respect to fires in its fossil-fired plants has been quite favorable. In operating 63 fossil-fired units at 12 major steam plants, TVA has experienced only two serious fires, one at the one-unit Bull Run Steam Plant in June of 1968, which resulted in a plant outage of eight weeks, and the other at Unit 7 of the Widows Creek Steam Plant in September of 1960, which resulted in an outage of six weeks. Of course, as in similar major industrial operations, there have been other minor fires. For example, in the four-year period from 1971 to present there were 17 such small fires, 15 of which involved only local damage to equipment and two of which, though more significant, did not result in more than a few days' outage.

These latter included a coal silo fire at Cumberland Steam Plant in September of 1973, and a crusher building fire at Shawnee Steam Plant in September of 1974.

The Bull Run fire was ignited when a fuel oil gasket failed allowing the fuel to spray on hot equipment. The resulting fire burned up through eight levels of the boiler bay, causing damage to several pieces of equipment such as lube oil pumps and ducting, as well as numerous cable trays which were in the bays. The fire was controlled and extinguished in a fairly short period of time using water. The Widows Creek fire involved the unit 7 boiler fuel pump-turbine in that a line flange separated and sprayed fuel oil on hot surfaces, causing ignition. Extensive damage resulted to boiler associated equipment and controls.

Our operating experience at nuclear power plants has not been sufficient to allow us to say that experience with fires at nuclear plants will be similar to that at our fossil-fired plants.

APPENDIX 11

QUESTIONS AND ANSWERS: JCAE/STATE OF ALABAMA

LETTER DATED SEPT. 17, 1975, MURPHY TO GODWIN, WITH ADDITIONAL QUESTIONS FOR STATE OF ALABAMA

RESPONSE BY ALABAMA DEPARTMENT OF HEALTH

JOINT COMMITTEE ON ATOMIC ENERGY,
U.S. CONGRESS,
Washington, D.C., September 17, 1975.

Mr. AUBREY GODWIN,
Division of Radiological Health,
State Health Department, Montgomery, Ala.

DEAR MR. GODWIN: Enclosed is the transcript of the September 16, 1975, hearing on the Browns Ferry fire. You should review and make any necessary corrections of your testimony in accordance with the instructions attached to the transcript. Also enclosed is a list of additional questions on your testimony which the Committee would like you to provide answers to for the record.

Since we are endeavoring to publish and distribute the hearing as rapidly as possible, it would be appreciated if you would return the corrected transcript and your answers to the additional questions by no later than September 30, 1975.

Thank you for your cooperation in this matter.

Sincerely yours,

GEORGE F. MURPHY, Jr.,
Executive Director.

Enclosures:

QUESTIONS FOR AUBREY GODWIN (STATE OF ALABAMA), BROWNS FERRY HEARING, SEPTEMBER 16, 1975

1. What are the responsibilities of the Division of Radiological Health (of the State Department of Health) with respect to the Browns Ferry Nuclear Plant?

2. Do you have any review or approval functions with regard to any aspect of nuclear power plant siting, design, construction or operation in the State of Alabama? Do you believe you should have such functions?

3. In your opinion, did the State receive adequate and timely information from the TVA regarding the fire so that the State could effectively monitor the situation and take any corrective action necessary?

4. The NRC report also indicates that, during the course of the fire and upon being informed of the status of events, the Division of Radiological Health concluded that the core cooling system was degraded and must be watched. To whom were these observations made, and what actions were taken as a result?

5. What actions, if any, does the State plan to take as a result of this accident? Are any changes to emergency planning or radiological monitoring procedures contemplated?

6. According to the NRC report (IV-13) some consideration was to be given to requesting additional appropriations from the State to increase surveillance activities around reactor sites. Has any action been taken with respect to this recommendation?

[Response dated September 25, 1976 subsequently received follows:]

Answers to Questions of the Joint Committee on Atomic Energy

1. What are the responsibilities of the Division of Radiological Health (of the State Department of Health) with respect to Browns Ferry Nuclear Plant?

Response:

Alabama Law assigns the State Board of Health (Alabama Department of Public Health) the following responsibilities:

1. Regulatory control of all sources of ionizing radiation consistent with Federal Regulatory Control of byproduct, source and special nuclear material.

2. During emergencies the Alabama Department of Public Health shall issue such orders as necessary to protect the public health and safety from sources of ionizing radiation.

3. The Alabama Department of Public Health shall inform the Governor of all invasions or threatened invasions of disease.

In summary, control of ionizing radiation sources from the reactors on the Browns Ferry Nuclear Plant site is Federal; control of ionizing radiation sources off the Browns Ferry Nuclear Plant site, including discharges in excess of U.S. Nuclear Regulatory Commission limits is by the Alabama Department of Public Health.

2. Do you have any review or approval functions with regard to any aspect of nuclear power plant siting, design, construction, or operation in the State of Alabama?

Response:

Our legislative mandate requires us to review any source of ionizing radiation which may have a public health or safety impact, regardless of who licenses the use of the source. To carry out this mandate, we regularly participate in all licensing proceedings for nuclear power plants in Alabama. In these proceedings, if we have reason to believe that inappropriate siting criteria, design, construction or operation is proposed, we will oppose the application. Certain portions of the plant design, construction and operation not related to radiation control may be under the regulatory control of other Divisions of the Alabama Department of Public Health or other Agencies of the State of Alabama.

Do you believe you should have such functions?

If you are asking should a program similar to the Agreement State program be established for nuclear power plants, my answer is NO.

The following are my reasons:

A. No state agency presently has an adequate staff to do the necessary multi-discipline review of a nuclear power plant. To acquire the necessary staff, a state would have to spend substantial sums of already short funds.

B. A given state will likely have only 1 4 sites and may get "stale" in its review of the operations of such a small number of facilities.

If you are asking should a state review these functions and take a position, my answer is YES.

I suggest that the following should be considered by the Joint Committee on Atomic Energy:

Those matters adjudicated as a part of a state process should not be readjudicated by the U.S. Nuclear Regulatory Commission, but should accept the sworn testimony and conclusions of the State Agency. This should not prevent the U.S. Nuclear Regulatory Commission from coming to a different conclusion from the State Agency on the overall project. However, the U.S. Nuclear Regulatory Commission should not be able to overrule a State Agency denial of a certificate of necessity or polution discharge permit. This should shorten the current hearing process, yet permit the hearing board to make the determinations required by NEPA.

3. In your opinion, did the State receive adequate and timely information from the TVA regarding the fire so that the State could effectively monitor the situation and take any corrective action necessary?

Response:

I do not feel that the State of Alabama was informed of the fire as promptly as we desired. Subsequently, we have clarified our position on notification by the TVA to include, not only radiation related events, but any event requiring emergency assistance by "off site" agencies. Once notified, the State was properl informed of this situation.

4. The NRC report also indicates that, during the course of the fire and upon being informed of the status of events, the Division of Radiological Health concluded that the core cooling system was degraded and must be watched. To whom were these observations made, and what actions were taken as a result?

Response:

These conclusions were given to:

A. Mr. W.T. Willis, Director
Environmental Health Administration
Alabama Department of Public Health

B. Ira L. Myers, M.D.
State Health Officer

The actions taken included:

A. The activation of the Alabama Radiation Emergency Plan

B. The activation of air sampling stations near Browns Ferry Nuclear Plant

C. The notification of Governor George C. Wallace

D. The close attention given to the current status of the reactors during each contact with the TVA

5. What actions, if any, does the state plan to take as a result of this accident? Are any changes to emergency planning or radiological monitoring procedures contemplated?

Response:

The State has taken the following actions as a result of its review of this accident:

A. Check and update the telephone directory at least every six months. The latest amendment to the Alabama Radiation Emergency Plan containing such changes is dated September 1, 1975.

B. Require each recipient of the Alabama Radiation Emergency Plan to document filing of changes. This procedure started with the September 1, 1975, amendment.

C. Now included in the Alabama Radiation Emergency Plan is a statement defining state and local responsibilities for informing officials of the Plan.

D. Arrangements have been made with the TVA to notify this Agency any time emergency assistance is requested of an "off-site" agency. This is not a part of the Alabama Radiation Emergency Plan since these calls will not be about a radiological incidnet. Potential radiological incidents will remain a portion of the Alabama Radiation Emergency Plan.

6. According to the NRC (IV-13) some consideration was to be given to requesting additional appropriations from the state to increase surveillance activities around reactor sites. Has any action been taken with respect to this recommendation?

Response:

The Division of Radiological Health's budget for the year ending September 30, 1975 consisted of the following sources of funds:

General Fund	$65,510.00
Transferred to the Division from funds appropriated for general public health use--Cigarette Tax	36,490.00
U.S. Nuclear Regulatory Commission contract for reactor monitoring	6,000.00
Total	$120,000.00

The Alabama Department of Public Health proposed in its 1976 budget the following:

General Fund		$159,868.00
To be transferred to the Division from funds appropriated for general public health use--Cigarette Tax		12,000.00
U.S. Nuclear Regulatory Commission contract for reactor monitoring		6,500.00
	Total	$178,368.00

The budget bill HB No. 490 in the current substitute proposes for the Division the following:

General Fund		$35,510.00
To be transferred to the Division from funds appropriated for general public health use--Cigarette Tax		36,490.00
U.S. Nuclear Regulatory Commission contract for reactor monitoring		6,500.00
	Total	$128,500.00

The Department of Public Health requested the introduction of a fee bill which would permit the charging of fees for radioactive material licenses and the the registration of sources of radiation. This is HB No. 967.

The projected revenue from this bill is $158,000 of which $22,000 would not be available to the Division. These funds would replace the Cigarette Tax funds and would allow for equipment expenditures not proposed in the original Departmental Budget.

This bill is currently pending before the House of Representatives of the Legislature.

APPENDIX 12

AEC BULLETINS ON USE OF POLYURETHANE

Issue No. 170

Issue No. 254

Issue No. 336

HEALTH and SAFETY
information
United States Atomic Energy Commission

ISSUE NO. 170 September 10, 1963

FOAMED POLYURETHANE FIRE PROVES DIFFICULT TO CONTROL

While the incident described below did not occur at an AEC facility, the lessons learned are of interest at AEC locations utilizing foamed polyurethane.

The incident occurred in a sphere (diameter 41') containing a 12'x 12'x 10' room designed to provide a very high degree of sound absorption through the use of numerous large pieces of foamed polyurethane which were individually shaped in the form of pyramids and bonded to an external metal framework. Eleven hours following the conduction of a test, this room was entered and nothing unusual was reported to have been noted. Approximately five hours later, when the room was next entered, smoke was noted coming from two of the pyramids. These were removed and work on modifying the ceiling of the room continued. From the details of the ensuing events given below, it is apparent that what was initially thought to be a minor fire problem turned out to be a complex firefighting task that could not be solved by application of large quantities of inert gas or thousands of gallons of water.

Work continued with two employees on top of the room walls drilling holes in angle iron support frames with portable electric drills. Smoke became evident in the room and fire in another part of the structure. One employee attempted to remove the burning pyramids while the fire department was being called. Smoke conditions were such that, without breathing equipment, the efforts to remove the burning materials had to be abandoned.

Upon arrival at 12:55 p.m., the fire department found light smoke coming out of the top and bottom of the sphere. The technician in charge requested that water be avoided, if possible, because of the possible damage it would do to electronic test equipment in the setup. An attack on the fire was made with CO_2 extinguishers which had little or no effect. Smoke increased until visibility was less than one foot.

A booster line was taken into the bottom opening of the sphere. Firefighters donned self-contained breathing apparatus and attacked the fire with water, which seemed to be controlling the surface fires. Visibility by this time was zero. A 2½" line was laid from the nearest hydrant (500' distant) and a 750 gpm pumper was placed in operation. This line was wyed into two 1½" lines, which were taken to the top opening of the sphere.

A blower was placed in the lower opening in an effort to force out some of the smoke and improve visibility. The water applied from the top was able to reach two sides of the room; however, this knocked down burning insulation to provide a heavy bed of glowing coals in the bottom of the sphere.

For approximately two hours this condition existed: a 1½" fog nozzle was used at the top opening discharging 90 gpm; a 1½" fog nozzle at the same location was used to protect the plyboard structure; and a 1½" line was operated through the bottom opening. Firefighters on top were able to operate without breathing equipment, but no entry through the bottom was possible without complete respiratory protection. Water was being drained out through plugs that were removed from the bottom of the sphere and was running out of the access hole where the attack was being made on the fire. It was evident that the overall temperature of the structure was being kept to a safe level but the water was doing nothing to extinguish fires within the pyramids.

The top hose was shut down and an exhaust fan placed over the opening to reduce smoke within the structure. Smoke conditions improved and firefighters were able to pick out hotspots and remove burning panels. A pole was used to break open the pyramids and improve accessibility to the deep-seated fires. A fire watch was posted until 9:00 p.m. when security was turned over to the guard force.

During the fire, approximately 75 tanks of breathing air, at least 24,000 gallons of water, and considerable quantities of carbon dioxide were used.

This incident pointed out the need for:

(a) A better appreciation of polyurethane fire risks by all personnel involved.

(b) Smoke-ejecting equipment and additional portable breathing equipment.

(c) Prompt notification of the fire department in case of <u>any</u> small fire.

(d) Consideration of attaching polyurethane so that it can be removed more easily in case of fire. (Note: No method is currently known for effectively controlling deep-seated fires within large pieces of foamed polyurethane.)

Division of Operational Safety
U. S. Atomic Energy Commission
Washington 25, D. C.

HEALTH and SAFETY
information
United States Atomic Energy Commission

ISSUE NO. 254 July 14, 1967

RECENT FOAMED POLYURETHANE FIRES ON NON-AEC FACILITIES

Incident #1

While cutting off bolts on the outside of an enclosed 20' x 20' x 80' test facility, two hot bolts and some molten metal fell (illustration 1) and lodged between the exterior aluminum wall cladding and the interior sprayed-on polyurethane insulation. Shortly thereafter, a firewatch smelled smoke and, after picking up an extinguisher, ran into the building but was immediately driven off by the intense heat and smoke. Smoke coming from the top of the tower attracted attention of fire department personnel who responded without waiting for receipt of a fire alarm. Intense smoke and heat prevented entrance through any opening into the building interior. Conduct of effective firefighting, even by personnel using self-contained respirators, was impossible. Hose streams applied to the external top of the building were ineffective. During the 20-minute duration of the blaze, periodic explosive sounds were heard, followed by puffs of smoke and fire.

Illustration 2 shows the point, 16' above floor level, at which hot bolts penetrated into the building interior to initiate a "chimney" fire which, as shown in illustrations 3 and 4, extended upward about 4'. Above this point, the blaze fanned out both vertically and horizontally to consume virtually all of the foamed polyurethane fuel in its path (see illustration 5) while leaving lower building wall portions unaffected.

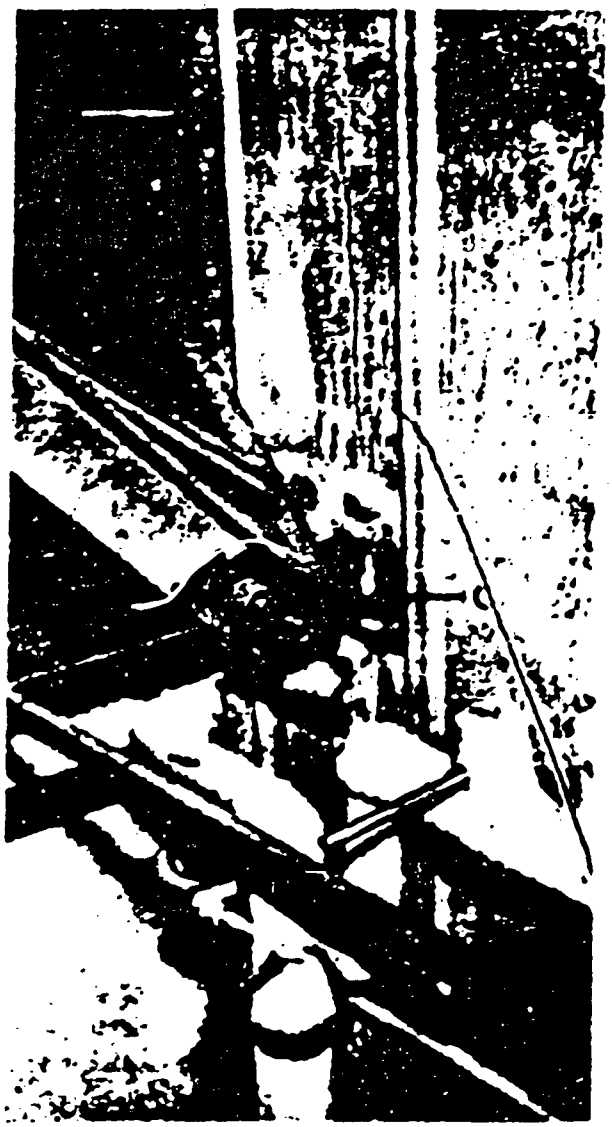
We understand that subsequent foamed polyurethane tests demonstrated ability to ignite test specimens with an acetylene torch but that self-sustaining combustion of the urethane did not persist after removal of the torch. Red hot bolts placed on other test specimens reportedly produced smoke and charring but no flame. These (and other) tests by the company were interpreted as showing that the polyurethane insulation did possess fire-resistant properties, but that under particular conditions of flame or sparks, and with the proper amounts of air, the material was proved capable of supporting and sustaining combustion. It was concluded that polyurethane insulation should not be used where there is a possibility of an open flame and spark and where a limited amount of air is present but that the insulation could be used where

(over)

- Illustration 1 -

Location "A" indicates the position of the two one-inch carbon steel bolts which were burned off the support walkway.

Location "B" indicates the point of penetration into the tower.



- Illustration 2 -

Location "C" is a closeup shot of the area indicated as Location "B" in illustration 1.

(continued)

- Illustration 3 -

Indicates the "chimney" where the fire started within the tower. The area behind the I-beam is a corner.

- Illustration 4 -

Indicates results of the fire behind the I-beam depicted in illustration 3. Fire burned in the "chimney" approximately four or five feet vertically before horizontal spread occurred.

- Illustration 5 -

Indicates completeness of burning. Note that only a small amount of urethane insulation remains after the fire and is charred.

(continued)

an infinite amount of air, or no air, was present. We understand that this conclusion will be implemented by applying foamed polyurethane to the building exterior but that in doing so extensive care will be taken to avert void areas, frequent noncombustible firebreaks will be provided, and a fire-resistant coating will be applied to the externally exposed portions of the insulation. (Note: While the conclusion concerning accentuated polyurethane combustibility under conditions involving restricted access to air was found surprising, it is of interest to note that all reports of polyurethane fire known to this office have uniformly stressed the profuse generation of dense smoke, a fact suggesting credibility of combustion under conditions limiting fire access to oxygen. It is suspected by many that the heat insulating properties of the foam may also play a major role in influencing urethane combustion.)

Incident #2

This incident occurred within a reactor containment enclosure. Smoke was noted about three hours after foamed polyurethane insulation had been sprayed onto some piping. Identification of the fire source was seriously impeded by the high density of the smoke. Using self-contained respirator equipment, it was found that the recently applied insulation was smoldering and was the source of "considerable smoke and soot." Roughly two hours after its detection, and after applying the contents of a large portable extinguisher, tripping a built-in CO_2 fire extinguishing system, and application of water sprays, the fire was extinguished. Subsequent investigation revealed fire damage was confined to insulation on a few pipelines, but smoke and soot damage was much more extensive.

This fire was attributed to spontaneous ignition resulting from the use of improper techniques in applying polyurethane foamed insulation. One report on the incident stated:

> "The curing process of the sprayed polyurethane insulation is exothermic. If applied in successive layers without sufficient curing time, its temperature will increase, breaking down its structure, and eventually causing ignition."

Following the incident, it is understood that all remaining polyurethane insulation was removed from the affected facility and replaced with materials considered to be less hazardous.

(over).

ISSUE NO. 254 July 14, 1967

Comments

The problem of controlling polyurethane fire was previously emphasized in HEALTH AND SAFETY INFORMATION Issue #170. Information concerning polyurethane fire risks is currently being gathered from a variety of sources. Analysis of this information is complicated by clear disagreement as to acceptable yardsticks for testing and evaluating these risks. Such conflicts are not surprising when it is recalled that many of the above-referenced tests, made following incident #1, could individually have been construed as providing evidence attesting that incident #1 could not occur. It is anticipated that a summary of findings resulting from study of urethane fire risks will be published during the next few months.

Division of Operational Safety
U. S. Atomic Energy Commission
Washington, D. C. 20545

GPO 925 777

ISSUE NO. 338
September 19, 1974

SERIOUS Accidents

THE UNITED STATES ATOMIC ENERGY COMMISSION

TWO FIRES INVOLVING URETHANE FOAM

The AEC recently experienced two fire losses in which polyurethane foam plastics played a major role. Following is a brief description of these fires, some background information on foams, and a summary of the lessons learned:

Occurrence Number 1

At a large laboratory, a number of accelerator beam tunnels of corrugated steel construction are connected to a 150- by 230-foot-experimental building. Near the far end of one 430-foot-long tunnel, workers were cutting a bolt hole with a welding torch from the outside of the structure. The tunnel, which is approximately 12-feet high, had been lined with a 1-1/2-inch thick, sprayed-on, polyurethane foam and during the cutting operation, this foam was ignited. Although the resulting fire revealed a number of fire protection deficiencies, only the problem of the foam is discussed here.

It was intended that the foam for this tunnel meet a flame spread rating of 25 and a smoke developed rating of 50 or less in accordance with UL 723 "tunnel test." The material that was actually installed was described as "self-extinguishing" by the ASTM 1692-67T test, but it did not have a "tunnel test" rating. In addition, it had been planned to coat the foam with an intumescent paint, install a 10-foot, glass fiber insulation firebreak at the tunnel end adjoining the experimental building and install a sprinkler water curtain at this end. At the time of the fire, the sprinklers were installed but valved off due to incomplete construction, the 10-foot firebreak was installed, and 150 feet of the foam closest to the experimental building were coated with a fire-retardant (but non-intumescent) paint.

Despite the above considerations, the fire swept the entire 430-foot length of the tunnel in less than 8 minutes, filled the laboratory with smoke, and burned plastic-reinforced fiberglass panels and plastic windows in the adjoining wall of the laboratory. The fire progressed rapidly with intense smoke development, similar to a test in a mine tunnel with dimensions which

Research Establishment in the March 1972 issue of the English magazine Fire. At the peak of the firefighting effort, over 2,000 gallons of water per minute were being applied to the fire by several fire departments.

Occurrence Number 2

A second incident occurred in a foundry-type building where urethane foam panels had been installed in a 5 by 17 by 13-foot-high breakout station to reduce noise levels. Lead-loaded vinyl curtains and plexiglass windows constituted the remaining combustibles. Occasionally, a leaking mold allows water to react with the hot metal, producing some pyrophoric residue and flammable gas which sometimes flashes when the mold is inverted in the breakout operation, resulting in a small fireball at the mouth of the pot. In this instance, the vinyl material and urethane ignited and forced evacuation of the building before the operators could take effective firefighting action. Dense smoke built up so fast that an ionization-type smoke detector some 260 feet south, in a separate room, alarmed within 5 minutes of the initial "flash."

Although both the curtain material and the foam were supposedly "self-extinguishing," tests showed the curtain material burned readily, producing large amounts of smoke, and the foam burned as long as it was exposed to flame. A test on a piece of older foam material showed that it ignited and burned freely, indicating an adverse aging effect.

Background on Foams

Small scale laboratory tests have long been recognized as inadequate to determine the fire properties of a material in large-scale applications. ASTM D635 and D1692 tests are used to compare relative flammabilities of plastic materials on a laboratory scale. Unfortunately, the terms "self-extinguishing" or "slow burning" as defined by these laboratory scale tests do not necessarily indicate a low combustibility in large scale wall and roof covering applications. The unsuitability of this interpretation was best demonstrated in an article in the NFPA quarterly for October 1962, wherein J. W. Wilson, Chief Construction Engineer for the Factory Mutual Engineering Division, utilized the tests to compare samples of four woods. Only balsa wood was classified as "burning" by the D635 test; all other woods were classified as "self-extinguishing." The further unsuitability of the D635 test for extrapolating into large scale applications is apparent from a review of 1971-72 Modern Plastics Encyclopedia where 146 of 149 plastics tested in accordance with D635 were listed as "self-extinguishing," "slow burning," etc.

(continued)

The application of the fire retardant paint to the surface of the foam in Occurrence 1 illustrates another difficulty in accepting test results without a full understanding of the application. While the material used was UL listed and FM approved, the applications applied to wood, masonry, plaster, and drywall surfaces. The application instructions did not cover the coating of foams. In fact, there is no fire retardant paint approved as a protective coating for foams.

The AEC Health and Safety Information Bulletin number 254 of July 14, 1967, reports two urethane foam fires which demonstrate another problem with these laboratory scale tests. While fire tests are performed by exposing the source of ignition to the exposed surface of the foam, fires have resulted from the ignition source applied to the outside, or protected face, or by a source, such as hot bolt, dropping between the metal wall and the insulating foam. This may be a more severe exposure than the normal test procedure since the heat may be concentrated by the insulating quality of the foam rather than being dissipated by the open air atmosphere which surrounds the normal test setup.

A summary of tests reported in the April 1973 issue of Fire concludes that flame retarded foams may be as easy to ignite as untreated foams when realistic ignition sources, such as a sheet of newspaper, are used. In fact, since the treated foams burned hotter, produced much denser smoke and at least as much carbon monoxide and hydrogen cyanide, it appears that a "fire retardant" type of foam could be a greater hazard in life safety situations than the untreated type.

The May 1974 issue of Fire Technology notes that the Urethane Safety Group and the Plastics in Construction Council, both of the Society of the Plastics Industry, have recently urged that the nation's building codes require that rigid urethane foam insulation be covered with a "suitable barrier against ignition" and not be left exposed after installation. The Urethane Safety Group is now preparing a new set of safety guidelines on applications which is expected to be available from the Society of the Plastics Industry, Department U-100, 250 Park Avenue, New York, New York 10017, in the near future. Factory Mutual Loss Prevention Data Sheets 1-57 and 8-29 are useful guides regarding methods of protecting foams. In the past, tests have been made on specific applications at AEC contractor sites and various guidelines were developed for specific situations or applications. In view of the current nationwide interest, the many tests now being made, and the new test methods under development, AEC criteria can be expected to change to reflect any new information.

Lessons Learned

While there are few significant new lessons to be learned from these two occurrences, a number of old lessons are evidently worth relearning. Specifically:

(over)

1. All new installations and modifications need a safety analysis--even modifications required for safety reasons. This is not the first instance in which a modification that was made to solve one safety problem created another.

2. The suitability of materials for any application must be based on the actual installation and its surrounding environment. Acceptance of fire tests in particular is not valid unless the tests apply to the intended installation. For example, a fire may be an acceptable risk while the smoke or toxic fumes which are generated may not be acceptable.

3. Merely citing that a material meets a test is meaningless without specifying the limits that are acceptable to the user. Claims that a product "meets a test" may be misleading in that tests merely standardize how a test is performed, not what results are acceptable. A statement that a plastic "meets ASTM E-84" (the tunnel test) is no more a guarantee of fire safety than a statement that a concrete "meets ASTM C-143" (the slump test) is a guarantee of quality. In both cases the user must specify the limit that is acceptable for his application.

4. Control of construction programs should consider sequencing of hazardous operations. Cutting and welding operations should preferably be completed before any necessary combustibles are introduced to the building. If this is not practical, it is preferable to delay the hazardous work until the protection systems are in service.

5. Fire protection systems should be in service before the building or equipment they protect is in service.

Division of Operational Safety
U. S. Atomic Energy Commission
Washington, D. C. 20545

APPENDIX 13

CORRESPONDENCE: NRC/COMEY

EXCHANGE OF CORRESPONDENCE BETWEEN NRC AND DAVID COMEY MARCH 24, 1975 THROUGH AUGUST 4, 1975

BUSINESS AND PROFESSIONAL PEOPLE FOR
THE PUBLIC INTEREST (PBI),
Chicago, Ill., March 24, 1975.

[Mailgram]

WILLIAM ANDERS,
Chairman, U.S. Nuclear Regulatory Commission,
Washington, D.C.:

Copy of mailgram sent this date to Bernard C. Rusche, Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

DEAR DR. RUSCHE: The March 22 occurrences in Dockets 50-259, 50-260, and 50-296 represent a serious failure to meet the intent of criterion 22, Appendix A, 10CFR 50. Therefore, pursuant to 10CFR 2.206, 2.202, and 50.100, BPI hereby formally requests you take the following specific actions.

(1) Review and identify those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems and show cause why these plants should not be required to shut down until they are retrofitted to meet said criteria.

(2) Adopt regulation requiring shutdown of operating units at multi-unit facilities during periods when work on a unit under construction could compromise the integrity of the engineered safety features on an operating unit or units.

DAVID DINSMORE COMEY,
Director of Environmental Research, BPI.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 8, 1975.

MR. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1901, Chicago, Ill.

DEAR MR. COMEY: This is to acknowledge receipt of your March 24, 1975 letter which refers to the recent fire at the Browns Ferry Nuclear Station and makes two requests for Commission action. These requests, filed under 10 CFR 2.206, are under active consideration and you may be assured that within a reasonable time, we will take the action appropriate in the circumstances.

The first request is being handled as a Petition for Order to Show Cause under 10 CFR 2.202, to operators of nuclear power plants "that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems". A copy of a related notice that is being filed with the Office of the Federal Register for publication is enclosed.

The second request is being handled separately as a petition for rulemaking. We plan to file a notice on this matter with the Office of the Federal Register for publication. A copy of our notice on your rulemaking petition will be sent to you by separate letter.

Sincerely,

EDSON G. CASE,
Acting Director,
Office of Nuclear Reactor Regulation.

Enclosure.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., April 8, 1975.

Docketing and Service Section, Office of the Secretary of the Commission.

PETITION FOR SHOW CAUSE ORDER

Two signed originals of an Order identified as follows are enclosed for your transmittal to the Office of the Federal Register for filing and publication: Nuclear Regulatory Commission, David D. Comey, Petition for Show Cause Order.

Twelve additional conformed copies of the Order are enclosed for your use.

A. SCHWENCER,
Light Water Reactors Branch 2-3,
Division of Reactor Licensing.

Enclosure.

NUCLEAR REGULATORY COMMISSION—DAVID D. COMEY

PETITION FOR SHOW CAUSE ORDER

Notice is hereby given that on March 24, 1975, David Dinsmore Comey made a request pursuant to 10 CFR § 2.206 of the Commission's Rules of Practice for the issuance of an Order to Show Cause why operation of nuclear powerplants should not be suspended pending full compliance with IEEE-279 criteria for nuclear powerplant protection systems. In accordance with the procedures specified in 10 CFR § 2.206, appropriate action will be taken on this request within a reasonable time.

A copy of the Petition for the Order to Show Cause is available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555.

Dated at Bethesda, Maryland this 8th day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION
EDSON G. CASE,
Acting Director,
Office of Nuclear Reactor Regulation.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., August 4, 1975.

Mr. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1901, Chicago, Ill.

DEAR MR. COMEY: This is in response to letter of June 16, 1975, which limits the request in paragraph (1) of your March 24 letter to a review by the NRC Staff of two plants (Indian Point 1 and Dresden 1) pending receipt of Dr. Hanauer's report on the investigation and review of the Browns Ferry fire. We are able to satisfy a portion of your request at this time.

For Indian Point 1, the Staff required the licensee in 1973 to provide information which compared the as-built plant protection system to IEEE-279, as published in 1968. Based on our review of that information, we determined that certain changes in the plant would provide substantial additional protection which is required for public health and safety. Accordingly, on June 11, 1973, the licensee was ordered, pursuant to 10 CFR 50.100, to complete these modifications within one and one-half years. Although these modifications have not yet been completed, the Indian Point 1 facility has been shut down since October 31, 1974. A copy of our June 11, 1973, order to the Consolidated Edison Company is enclosed for your information.

With regard to Dresden 1, we have recently requested Commonwealth Edison to compare the as-built design of the plant with the 1968 edition of IEEE-279, to identify features of the plant that do not comply with this standard, and to determine whether, based on this review, there are changes in the facility that should be accomplished which provide substantial additional protection which is required for public health and safety. A copy of our letter to Commonwealth Edison dated July 31, 1975 is enclosed. Based on our evaluation of the licensee's report, the staff will then determine what actions, if any, with regard to Dresden 1 are necessary to protect the health and safety of the public.

We expect that the licensee and Staff review will require several months to complete, and will provide you with the results of these reviews.

Sincerely,

EDSON G. CASE,
Deputy Director,
Office of Nuclear Reactor Regulation.

Enclosures:

ENCLOSURE 1

U.S. ATOMIC ENERGY COMMISSION,
Washington, D.C., June 11, 1973.

Attn: MR. WILLIAM E. CALDWELL, *Vice President,*
Consolidated Edison Co. of New York, Inc.,
4 Irving Place, New York, N.Y.

GENTLEMEN: As you know, we have under review your application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station. Our review includes consideration of the need for backfitting Unit No. 1 to provide substantial, additional protection which is required for the public health and safety. Our review of portions of the plant protection system has been completed, and we find that modifications to the system are necessary.

In the event of an accident jeopardizing the integrity of the core, Section 6 (Accident Analysis) of the Safety Analysis Report for Core B indicates that the plant protection system would function to terminate the accident before fuel and fuel clad melting could occur in the hottest channel and result in the release of fission products to the reactor coolant system. Although containment is provided to retain fission products should they be released from the reactor coolant system, protection systems for modern nuclear plants are designed to meet the single failure criterion of IEEE Std. 279–1968, "Criteria for Nuclear Power Plant Protection Systems", and the General Design Criteria of Appendix A to 10 CFR Part 50. By letters dated March 31, 1972, and April 28, 1972, from your attorneys, you submitted in support of your application additional information which compares the plant protection system to IEEE 279 as published August 30, 1968, and to General Design Criterion 20, "Protection Systems Redundancy and Independence", and General Design Criterion 21, "Single Failure Definition", as published in the *Federal Register* on July 11, 1967. Our review is based on this information and other information in the public record.

We find that improvements in the plant protection system, including circuits for isolating containment and for protection against steam line breaks, will provide substantial, additional protection for the health and safety of the public, and that such improvements are required for the health and safety of the public. Therefore, pursuant to Section 50.109, "Backfitting", of 10 CFR Part 50, you are hereby directed to submit, within six months from the date of this letter, the following:

1. Details of proposed modifications that are necessary to make the design of protection circuits essential to safety conform with the requirements of Section 4.2, "Single Failure Criterion", of IEEE Std. 279–1968.
2. Details of proposed modifications and analyses of such modifications that are required to assure that dc electric power systems essential to safety meet the single failure criterion.
3. Your detailed analyses of all portions of the protection system including the proposed modifications which show that the modified system meets the single failure criterion.
4. Results of analyses and/or tests which demonstrate that all instrumentation and electric equipment essential to safety can function in its environment during and following an accident. If satisfactory results are not obtained, describe the modifications necessary to assure that all instrumentation and electric equipment essential to safety can function in its accident environment.

Further, you are hereby directed to complete within one and one-half years from the date of this letter, all such modifications and tests necessary to conform to the requirements of the single failure criterion.

In developing the proposed modifications necessary to satisfy the requirements specified in 1. and 3. above, the following concepts should be applied in your analyses of the modifications.

1. The types of single failures to be considered are:
   a. failure to any single component or module or a single circuit fault. (Circuit faults include short circuits, open circuits, grounds, and the application of any available ac or dc potential.)

b. multiple component failures and/or circuit faults that can be predicted to occur as a result of a single event within or external to the system.

2. Each single failure is considered to occur coincident with any and all combinations of nondetectable failures. (Nondetectable failures are failures that cannot be detected by periodic tests, anomalous indications, or alarms.)

3. If an event that results in the need for protective action can cause failures in the protection system or if failures in circuits common to control and protective functions can result in the need for protective action, it must be shown that no single failure, in addition to the event-caused failures or the failures in the common portion of the protection system, will prevent the protection system from performing the required protective action(s).

Based on our review of portions of the protection system, the following suggested modifications are examples of the kind of modifications that might be made to the protection system in order to satisfy the single failure criterion:

(a) install an additional scram bus similar to the existing scram bus;

(b) connect the two scram buses to the scram solenoid valves in a manner which ensures that either bus is capable of deenergizing the solenoid valves regardless of whether the other bus is operable or failed;

(c) disconnect one of the nuclear instrumentation logic networks from the existing scram bus and connect it to the additional similar scram bus;

(d) transfer either the high pressurizer pressure or the high primary pressure trip relay contacts from the existing scram bus to the additional similar scram bus;

(e) install additional independent high primary temperature, low primary pressure, and low boiler feedwater pressure instrumentation for use with the additional similar scram bus;

(f) transfer one flux/flow computer from the existing scram bus to the additional similar scram bus and install additional flux averaging amplifiers, one for use with the additional similar scram bus and one for use with the automatic reactor control system;

(g) modify the existing scram bus and design the additional similar scram bus (or design their connections to the scram solenoid valves) in a manner that permits testing individually the two scram buses during reactor power operation; and

(h) provide redundant initiating devices in the criteria for protecting against the effects of a steam line break and for isolating containment.

If we can provide additional guidance with regard to your preparation of these required submittals, please do not hesitate to call (Mr. Roger Woodruff is the Regulatory Project Manager).

You should appreciate that, in addition, there are other technical issues related to your pending application for a full term operating license for Indian Point Unit No. 1 now under review by the AEC staff. In this connection, we are concerned with the frequent delays that have been experienced in your responses to our requests for additional information on these issues. We believe that this matter warrants prompt management attention and request that you take the necessary steps to remedy this situation.

Sincerely,

A. GIAMBUSSO,
Deputy Director for Reactor Projects,
Directorate of Licensing.

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING—CONSOLIDATED EDISON CO. OF NEW YORK, INC.

INDIAN POINT 1; DOCKET NO. 50-3

Background

As part of our review of Consolidated Edison Company's application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station, we have completed a review to determine the need, if any, for backfit of the plant protection system. Our position and the reasons for it are discussed in the following sections.

Description of the Plant Protection System

Principal features of the Unit No. 1 plant protection system are a scram bus, two logic networks, and circuits for isolating containment and for isolating steam lines.

When values of important plant parameters exceed their limiting safety system settings, relays in the scram bus automatically cause insertion of the control rods by deenergizing a solenoid valve in the hydraulic system for each control rod. Principal relays are controlled by instrumentation which monitors reactor coolant temperature and pressure, boiler feedwater pressure, and the ratio of neutron flux to reactor coolant flow.

Neutron flux sensors monitor reactor power and provide input signals to the logic networks. These networks supply electrical power to the scram bus and are intended to interrupt electrical power to the scram bus if reactor power exceeds the limiting safety system setting.

Multiple input signals to the logic networks are provided by each of 18 neutron flux sensors. The signals are connected in sets of three to each of nine "two-out-of-three" modules in each network. If any two signals to a "two-out-of-three" module exceeds the limiting safety system setting, the module initiates a scram signal. The outputs of the nine "two-out-of-three" modules are connected to a "one-out-of-nine" module. On receipt of a scram signal, a solid state switch in this module "opens" interrupting power to two trip actuator amplifiers. A solid state switch in each of the parallel trip actuator amplifiers then "opens", disconnecting power from the scram bus. Because the two logic networks are connected to the scram bus in parallel, both logic networks must function to deenergize the scram bus.

Three of the 18 neutron flux sensors also provide signals to a flux averaging amplifier. The output from this amplifier and signals from reactor coolant flow sensors are connected to two computers which calculate the ratio of flux to flow. If the value of the ratio is excessive, each computer demands that each of two pairs of parallel relay contacts opens in the scram bus. Other pairs of parallel relay contacts in the scram bus are opened by sensors monitoring boiler feedwater pressure, reactor cooling system pressure, and reactor cooling system temperature.

To prevent overcooling of the reactor core in the event of a steam-line-break accident and to limit the reactivity and power transients which would result, circuitry is provided to close the steam stop valve in the affected loop if a reactor scram occurs in coincidence with low steam pressure. A pressure drop across the steam stop valve in coincidence with scram causes closure of the stop valves in the associated reactor cooling loop. These functions are initiated by a single switch or relay.

Containment is partially or completely isolated automatically in the event of high containment pressure, high radiation, and very low pressurizer level. For each case, isolation is initiated by a single relay.

Safety Evaluation

IEEE 279 establishes several criteria for the design of plant protection systems. However, the fundamental aspects of IEEE 279 is the single failure criterion which is intended to assure that the protection system will function in the event that any single failure or fault exists somewhere in the protection system. Our safety evaluation is based on this fundamental aspect of IEEE 279.

With regard to the scram bus, application of stray power to it can disable all relays and solid state switches from the point of application of stray power back to, and including, the trip actuator amplifiers, thus, defeating the scram functions associated with the relays and solid state switches. Other kinds of single failures, e.g., failed components, short circuits, open circuits, and grounds, can also disable relays in the scram bus and cause loss of specific scram functions.

In the logic networks, random failure of any one of six solid state switches in the "closed position" will disable the high neutron flux scrams. In the circuits for isolating containment and for isolating steam lines, single failures of initiating mechanisms would cause loss of protective action.

Although we have not reviewed all portions of the plant protection system, it is clear that single failures can be expected to inhibit the plant protection system and that backfitting the plant protection system will provide substantial, additional protection which is required for the health and safety of the public.

Backfit and Information Requirements

Pursuant to 10 CFR Part 50, Section 50.109, we are requiring that Consolidated Edison submit, within six months from the date of this evaluation, proposed modifications to the plant protection system which will provide significantly greater assurance that the plant protection system will perform its intended function should any of the following kinds of single failures occur: failed components, short circuit, open circuits, grounds, and application of any available ac or dc potential. The proposed modifications must include capability at power

for conducting surveillance on the plant protection system to assure that it is capable of performing its intended function. Further, we are requiring that Consolidated Edison complete approved modifications of the plant protection system within one and one-half years from the date of this evaluation.

R. W. WOODRUFF,
Operating Reactors Branch #1,
Directorate of Licensing.
ROBERT J. SCHEMEL,
Operating Reactors Branch #1,
Directorate of Licensing.

U.S. NUCLEAR REGULATORY COMMISSION,
Washington, D.C., July 31, 1975.

Attention Mr. J. S. Abel.
COMMONWEALTH EDISON CO.,
Nuclear Licensing Administrator, Boiling Water Reactors,
P.O. Box 767, Chicago, Ill.

DEAR MR. ABEL: As you may know, on March 24, 1975, Mr. David Comey, representing Businessmen for the Public Interest, requested, among other things, that the NRC identify those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems, and show cause why such plants should not be required to shut down until they are retrofitted to meet these criteria. A copy of Mr. Comey's request is enclosed (Enclosure 1). Based on a review of information in our files, those plants which did not comply with IEEE-279 were identified and provided to Mr. Comey in a letter dated May 5, 1975 (Enclosure 2). In this letter, we suggested to Mr. Comey that BPI might wish to modify its show cause request pending completion of the NRC's ongoing investigation and review of the Browns Ferry fire. In his reply of June 16, 1975, Mr. Comey agreed to such a modification, except as his initial request related to Indian Point 1 and Dresden 1. A copy of Mr. Comey's June 16 letter is also enclosed (Enclosure 3).

With regard to the Indian Point 1 facility, the NRC staff has previously conducted a review of the design of this facility with regard to compliance with IEEE-279 and ordered the licensee pursuant to 50.109 to make certain modifications to the plant. A copy of our letter to Consolidated Edison dated June 11, 1973, concerning this matter is enclosed for your information (Enclosure 4). With regard to Dresden 1, the information available to the staff is not sufficient to make a comparable evaluation. Accordingly, pursuant to 50.109(c) of the Commission's regulations, you are requested to compare the as-built design of the Dresden 1 plant with the 1968 edition of IEEE-279, to identify features of the plant that do not comply with this standard, and to determine whether, based on your evaluation, there are changes in the facility that should be accomplished which provide substantial additional protection which are required for public health and safety. The results of your evaluation should be provided to the staff within 60 days of the date of this letter.

If you have any questions concerning this matter, please let me know.

Sincerely,

EDSON G. CASE, *Deputy Director,*
Office of Nuclear Reactor Regulation.

Enclosures:

ENCLOSURE 1

BERNARD C. RUSCHE,
Director, Office of Nuclear Reactor Regulations, U.S. Nuclear Regulatory Commission, Washington, D.C.

DEAR DR. RUSCHE: The March 22 occurrences in dockets 50-259, 50-260, and 50-296 represent a serious failure to meet the intent of criterion 22, Appendix A, 10 CFR 50, therefore, pursuant to 10CFR 2.206, 2.202, 50.100, BPI hereby formally requests you take the following specific actions.

(1) Review and identify those operating nuclear power plants that not fully comply with IEEE-279 criteria for nuclear power plant protection systems and show cause why these plants should not be required to shut down until they are retrofitted to meet said criteria.

(2) Adopt regulation requiring shutdown of operating units at multi-unit facilities during periods when work on a unit under construction could com-

promise the integrity of the engineered safety features on an operating unit or units.

DAVID DINSMORE COMEY,
Director of Environmental Research, BPI.

ENCLOSURE 2

MAY 5, 1975.

Mr. DAVID DINSMORE COMEY,
Director of Environmental Research, Business and Professional People for the Public Interest, 109 North Dearborn Street, Suite 1001, Chicago, Ill.

DEAR MR. COMEY: Your March 24 letter asked, among other things, that those operating nuclear power plants that do not fully comply with IEEE-279 criteria for nuclear power plant protection systems be identified. As you know, the IEEE-279 criteria were initially published in August 1968 (as "Proposed Criteria"), and have been revised once since that time (in 1971).

Enclosure 1 is a list of those operating plants that have been evaluated by the NPC staff for compliance with the IEEE-279 criteria and found to be acceptable, and the date of issuance of the applicable IEEE criteria. Enclosure 2 lists the remaining eight operating plants that were evaluated for compliance against earlier staff criteria available at the time those plants were reviewed and approved for operation. These eight plants were all licensed prior to August 1968. Although some of the earlier criteria used by the staff in its evaluations of the reactor protection systems for the plants listed in Enclosure 2 were subsequently included in what is now IEEE-279, the degree of compliance with the 1968 or 1971 IEEE-279 criteria varies for these plants.

Your March 24 letter also requested, in view of the Browns Ferry fire, that the NRR review those plants that do not fully comply with the IEEE-279 criteria for nuclear power plant protective systems, and show cause why these plants should not be required to shut down until they are retrofitted to meet these criteria. As I discussed with you by telephone, such a review by NRC would initially involve a report by the involved licensees dealing with the issue that you have raised, and a subsequent evaluation and judgment by the staff. I would expect that these steps would take at least six months to complete and would involve considerable staff effort.

Pending completion of the ongoing NRC investigation and review of the cause and implications of the Browns Ferry fire, the NRC has taken actions through its Office of Inspection and Enforcement that we believe provide reasonable assurance that all operating plants, including those listed in Enclosure 2, can continue to operate without undue risk to the health and safety of the public. These actions are set forth in Enclosures 3 and 4 to this letter, and were taken in late March and early April of this year.

The NRC actions taken to date as a result of the Browns Ferry fire deal primarily with measures to reduce the likelihood of occurrence of serious fires at operating nuclear power plants. A related important issue is whether the criteria (such as those in IEEE-279) relevant to limiting the damage from such a fire, if one should occur, should be revised in light of the Browns Ferry occurrence. For this reason, one of the specific objectives of the ongoing NRC investigation and review is to determine whether or not such criteria should be modified or supplemented and, if so, to which plants these revised or supplemental criteria should be applied. The results of the Browns Ferry investigation by NRC's Office of Inspection and Enforcement are presently expected to be publicly available in July 1975. The schedule for completion of the technical review by the group under the direction of the Technical Advisor to the Executive Director for Operations has not yet been established.

In view of the foregoing, it appears possible that you may wish to modify the request in paragraph (1) of your letter of March 24. For example, you might wish to consider deferral of any further action by NRC on this portion of your request until the NRC's investigation and review of the Browns Ferry fire is completed, limiting this portion of your request to only some of the plants listed in Enclosure 2, or other alternative courses of action.

I would be pleased to discuss this matter with you at your convenience before we continue action on your request. If you need any further information, please let me know.

Sincerely yours,

EDSON G. CASE,
Deputy Director, Office of Nuclear Reactor Regulation.

Enclosures:

OPERATING NUCLEAR POWER REACTORS THAT WERE EVALUATED AGAINST THE CRITERIA CONTAINED IN IEEE STD 279 AND FOUND ACCEPTABLE

Docket No.	Plant	Applicable issue of IEEE-279	Operating license issuance date
50-219	Oyster Creek	1968	Apr. 9, 1969
50-220	Nine Mile Point 1	1968	Dec. 26, 1974
50-237	Dresden 2	1968	Dec. 22, 1969
50-244	Ginna	1968	Sept. 19, 1969
50-245	Milestone Point 1	1968	Oct. 7, 1970
50-247	Indian Point 2	1968	Oct. 19, 1971
50-249	Dresden 3	1968	Jan. 12, 1971
50-250	Turkey Point 3	1968	July 19, 1972
50-251	Turkey Point 4	1968	Apr. 10, 1973
50-254	Quad Cities 1	1968	Oct. 1, 1971
50-255	Palisades	1968	Mar. 24, 1971
50-259	Browns Ferry 1	1968	June 26, 1973
50-260	Browns Ferry 2	1968	June 24, 1974
50-261	H. B. Robinson 2	1968	July 31, 1970
50-263	Monticello	1968	Sept. 8, 1970
50-265	Quad Cities 2	1968	Mar. 21, 1972
50-266	Point Beach 1	1968	Oct. 5, 1970
50-267	Fort St. Vrain	1968	Dec. 21, 1973
50-269	Oconee 1	1968	Feb. 6, 1973
50-270	Oconee 2	1968	Oct. 6, 1973
50-271	Vermont Yankee	1968	Mar. 21, 1973
50-277	Peach Bottom 2	1968	Aug. 8, 1973
50-278	Peach Bottom 3	1968	July 2, 1974
50-280	Surry 1	1968	May 25, 1972
50-281	Surry 2	1968	Jan. 29, 1973
50-282	Prairie Island 1	1968	Aug. 9, 1973
50-285	Fort Calhoun	1968	May 24, 1973
50-287	Oconee 3	1968	July 19, 1973
50-289	Three Mile Island 1	1968	Apr. 19, 1974
50-293	Pilgrim 1	1968	June 8, 1972
50-295	Zion 1	1968	Oct. 19, 1973
50-298	Cooper Station	1968	Jan. 18, 1974
50-301	Point Beach 2	1968	Nov. 16, 1971
50-304	Zion 2	1968	Nov. 14, 1973
50-305	Kewaunee	1968	Dec. 21, 1973
50-306	Prairie Island 2	1968	Oct. 29, 1974
50-309	Maine Yankee	1968	Sept. 15, 1972
50-312	Rancho Seco 1	1968	Aug. 16, 1974
50-313	Arkansas One, Unit 1	[1] 1971	May 21, 1974
50-315	D.C. Cook 1	1968	Oct. 25, 1974
50-317	Calvert Cliffs 1	1968	July 31, 1974
50-321	Hatch 1	1968	Aug. 6, 1974
50-324	Brunswick 2	[1] 1971	Dec. 27, 1974
50-331	Duane Arnold	1968	Feb. 22, 1974
50-333	Fitzpatrick	[1] 1971	Oct. 17, 1974
OPERATING NUCLEAR POWER REACTORS THAT WERE EVALUATED AGAINST PRE IEEE 279 CRITERIA			
50-003	Indian Point 1		Mar. 26, 1962
50-010	Dresden 1		Sept. 28, 1959
50-029	Yankee Rowe		Dec. 24, 1963
50-133	Humboldt Bay		Aug. 28, 1962
50-155	Big Rock Point		Aug. 30, 1962
50-206	San Onoire 1		Mar. 27, 1967
50-213	Connecticut Yankee		June 30, 1967
50-409	LaCrosse		July 3, 1967

[1] The differences between the 1968 and 1971 version of IEEE-279 do not appear.

CABLE FIRE AT BROWNS FERRY NUCLEAR POWER STATION—IE BULLETIN NO. 75-04

DESCRIPTION OF CIRCUMSTANCES

Preliminary information from Tennessee Valley Authority regarding a fire which occurred on March 22, 1975, at their Browns Ferry site near Athens, Alabama, indicates that the fire was started as a result of construction activities. The fire resulted in the shutdown of two operating nuclear plants and made several safety systems inoperative, including systems normally used for decay heat removal during shutdown. The workmen were engaged in construction activities on a third unit not yet licensed for operation by NRC.

Initial information indicates that during the installation and testing of cable through-wall penetrations an open flame ignited a flammable material used in the penetration seals.

ACTION TO BE TAKEN BY LICENSEES:

The following actions are requested of selected Licensees with operating power reactor facilities and major construction activities at a common site:

1. Review your overall procedures and system for controlling construction activities that interface with reactor operating activities, with particular attention to the installation and testing of seals for electrical cables between compartments of the reactor building, e.g., control room to cable spreading room.
2. Review the design of floor and wall penetration seals, with particular attention to the flammability of materials.
3. Evaluate your procedures for the control of ignition sources which may be used for leak testing or other purposes in areas containing flammable materials.
4. Report to this office, in writing within 20 days of the date of this Bulletin, the results of your reviews or evaluations regarding items 1 through 3 above.

ACTIONS TO BE TAKEN BY LICENSEES MAY BE REVISED:

The actions requested by Licensees above may be revised as additional details of the Browns Ferry occurrence are available and evaluated by the NRC.

CABLE FIRE AT BROWNS FERRY NUCLEAR PLANT—IE BULLETIN No. 75-04A

The following material supplements and modifies IE Bulletin 75-04.

DESCRIPTION OF CIRCUMSTANCES

Additional, though still preliminary, information has become available related to the fire which occurred at the Browns Ferry Site on March 22, 1975. The fire started in the cable spreading room at a cable penetration through the wall between the cable spreading room and the reactor building for Unit 1. A slight differential pressure is maintained (by design) across this wall, with the higher pressure being on the cable spreading room side. The penetration seal originally present had been breached to install additional cables required by a design modification. Site personnel were resealing the penetration after cable installation and were checking the airflow through a temporary seal with a candle flame prior to installing the permanent sealing material. The temporary sealing material was highly combustible, and caught fire. Efforts were made by the workers to extinguish the fire at its origin, but they apparently did not recognize that the fire, under the influence of the draft through the penetration, was spreading on the reactor building side of the wall. The extent of the fire in the cable spreading room was limited to a few feet from the penetration; however, the pressure of the fire on the other side of the wall from the point of ignition was not recognized until significant damage to cables related to the control of Units 1 and 2 had occurred.

Although control circuits for many of the systems which could be used for Unit 1 were ultimately disabled by the fire, the station operating personnel were able to institute alternative measures by which the primary system could be depressurized and adequate cooling water supplied to the reactor vessel. Unit 1 was shut down manually and cooled using remote manual relief valve operation and condensate booster pump, and control rod drive system pumps. Unit 2 was shut down and cooled for the first hour by the RCIC. After depressurization, Unit 2 was placed in the RHR shutdown cooling mode with makeup water available from the condensate booster pumps and control rod drive system pump.

ADDITIONAL ACTIONS TO BE TAKEN BY LICENSEES

1. Because the occurrence appears to have resulted from modifications being made to an operating unit, all power reactors with operating licenses should address the actions requested in Bulletin 75-04 as well as the actions described below.
2. Review your policies and procedures relating to construction or maintenance and modification work to assure that activities which might affect the safety of a unit in operation, including the ability to shut down and cool the unit are properly controlled. Your review should consider particularly your policy on deferring construction, maintenance or modification work on a unit

until a shutdown period except for emergency maintenance vital to continued safe operating or safe shutdown of the unit.

3. Review your policies and procedures to assure that for construction or modification and maintenance activities during plant operation, particular attention is given to the following areas:

(a) The degree of safety significance of affected and nearby cabling and piping.

(b) The use and control of combustible materials.

(c) The use and control of equipment that may be an ignition source.

(d) The assignment of personnel, knowledgeable of plant arrangement and plant operations, whose sole temporary responsibility is monitoring the safe performance of construction or maintenance and modification work, including attention to otherwise unattended areas adjacent to the work areas.

(e) Provision of installed or portable equipment to provide the monitoring personnel with prompt communication with the operating staff in the control room.

(f) Provision of adequate fire prevention and fire suppression equipment, installed or portable, for the following locations:

(1) Areas where work is being performed.

(2) Areas where occurrence of a fire has high safety significance, even though the probability of occurrence is relatively small.

(g) Recognition that fire, even one involving electrical equipment, may, if of sufficient intensity require water as the ultimate suppression medium.

4. Review your emergency procedures to assure that consideration for alternate methods for accomplishing an orderly plant shutdown and cooldown are provided in case of loss of normal and preferred alternative shutdown and cooldown systems for any reason (e.g. a fire). In this connection, assure that the minimum information necessary to assist the operators in such shutdown actions, the minium protection system actions required (e.g. scram) and the spectrum of alternative paths available to the operators to supply cooling water and remove decay heat dependent on plant conditions are included in your emergency procedures.

5. Report to this office, in writing, within 20 days of the date of this Bulletin, your schedule for review in each of the above areas.

6. Upon completion of your reviews provide the results of the review and the schedule for accomplishment of any revisions to your policies and procedures, and any proposed changes to the facility, and the date by which the changes are scheduled to be completed. If this latter date is more than 30 days after the date of the initial report, provide a monthly summary report detailing your progress in the review and/or proposed procedure or facility modifications. Reports requested by Bulletin 75-04 may be incorporated with the initial response to this Bulletin.

BUSINESS AND PROFESSIONAL PEOPLE FOR THE PUBLIC INTEREST,
Chicago, Illinois, June 16, 1975.

EDSON G. CASE,
Deputy Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C.

DEAR MR. CASE: Thank you for your letter of May 5, 1975, responding to my March 24, 1975, letter.

We have reviewed Enclosure 2 to your letter, listing those operating nuclear power reactors that were evaluated against pre-IEEE-279 criteria.

We have no desire to tie up NRC staff with a review of all the plants listed in Enclosure 2, and if the actions taken by NRC pending review of the cause and implications of the Browns Ferry fire will assure adequate protection of the public health and safety, and if the results of the Hananer review are going to be forthcoming soon, we are willing to modify the request in paragraph (1) of my letter of March 24, 1975.

There are, however, two plants listed in Enclosure 2 which we believe ought to be treated separately. These are the two oldest plants, Indian Point 1 and Dresden 1, the operating license issue date for which are March 26, 1962, and September 28, 1959, respectively. Because of their early operating license review dates, it appears likely that these two plants may have the least compliance

with IEEE-279. Moreover, of the eight reactors listed in your Enclosure 2, these two reactors are the closest to large metropolitan population centers (New York and Chicago, respectively).

In addition, both Indian Point 1 and Dresden 1 have rather poor operating records, both with respect to numbers of abnormal occurrences and also noncompliance with NRC regulations, and, therefore, we believe that it appears that there is a higher probability of a serious accident at these facilities than at the others listed in Enclosure 2.

Accordingly, as you suggest in the next to last paragraph of your letter of May 5, 1975, we will limit our request in paragraph (1) of my March 24 letter to a review by NRC, including a report by the involved licensees and an evaluation and judgment by the staff, of only some of the plants in Enclosure 2, namely, Indian Point 1 and Dresden 1. As you further suggest, we are willing to consider deferral of any further action by NRC with respect to other plants on that portion of our request until we have received the Hanauer investigation and review of the Browns Ferry fire.

Very truly yours,

DAVID DINSMORE COMEY,
Director, Environmental Research.

ENCLOSURE 4

U.S. ATOMIC ENERGY COMMISSION,
Washington, D.C., June 11, 1973.

Attn: Mr. WILLIAM E. CALDWELL, *Vice President,*
Consolidated Edison Co. of New York, Inc.,
4 Irving Place, New York, N.Y.

GENTLEMEN: As you know, we have under review your application for a full-term operating license for Unit No. 1 at the Indian Point Nuclear Generating Station. Our review includes consideration of the need for backfitting Unit No. 1 to provide substantial, additional protection which is required for the public health and safety. Our review of portions of the plant protection system has been completed, and we find that modifications to the system are necessary.

In the event of an accident jeopardizing the integrity of the core, Section 6 (Accident Analysis) of the Safety Analysis Report for Core—indicates that the plant protection system would function to terminate the accident before fuel and fuel clad melting could occur in the hot-test channel and result in the release of fission products to the reactor coolant system. Although containment is provided to retain fission products should they be released from the reactor coolant system, protection systems for modern nuclear plants are designed to meet the single failure criterion of IEEE Std. 279-1968, "Criteria for Nuclear Power Plant Protection Systems", and the General Design Criteria of Appendix A to 10 CFR Part 50. By letters dated March 31, 1972, and April 28, 1972, from your attorneys, you submitted in support of your application additional information which compares the plant protection system to IEEE 279 as published August 30, 1968, and to General Design Criterion 20, "Protection Systems Redundancy and Independence", and General Design Criterion 21, "Single Failure Definition", as published in the *Federal Register* on July 11, 1967. Our review is based on this information and other information in the public record.

We find that improvements in the plant protection system, including circuits for isolating containment and for protection against steam line breaks, will provide substantial, additional protection for the health and safety of the public, and that such improvements are required for the health and safety of the public. Therefore, pursuant to Section 50.109, "Backfitting", of 10 CFR Part 50, you are hereby directed to submit, within six months from the date of this letter, the following:

1. Details of proposed modifications that are necessary to make the design of protection circuits essential to safety conform with the requirements of Section 4.2, "Single Failure Criterion," of IEEE Std. 279-1968.

2. Details of proposed modifications and analyses of such modifications that are required to assure that dc electric power systems essential to safety meet the single failure criterion.

3. Your detailed analyses of all portions of the protection system including the proposed modifications which show that the modified system meets the single failure criterion.

4. Results of analyses and/or tests which demonstrate that all instrumentation and electric equipment essential to safety can function in its environment during and following an accident. If satisfactory results are not obtained, describe the modifications necessary to assure that all instrumentation and electric equipment essential to safety can function in its accident environment.

Further, you are hereby directed to complete within one and one-half years from the date of this letter, all such modifications and tests necessary to conform to the requirements of the single failure criterion.

In developing the proposed modifications necessary to satisfy the requirements specified in 1. and 3. above, the following concepts should be applied in your analyses of the modifications.

1. The types of single failures to be considered are:

a. failure to any single component or module or a single circuit fault. (Circuit faults include short circuits, open circuits, grounds, and the application of any available ac or dc potential.)

b. multiple component failures and/or circuit faults that can be predicted to occur as a result of a single event within or external to the system.

2. Each single failure is considered to occur coincident with any and all combinations of nondetectable failures. (Nondetectable failures are failures that cannot be detected by periodic tests, anomalous indications, or alarms.)

3. If an event that results in the need for protective action can cause failures in the protection system or if failures in circuits common to control and protective functions can result in the need for protective action, it must be shown that no single failure, in addition to the event-caused failures or the failures in the common portion of the protection system, will prevent the protection system from performing the required protective action(s).

Based on our review of portions of the protection system, the following suggested modifications are examples of the kind of modifications that might be made to the protection system in order to satisfy the single failure criterion:

(a) install an additional scram bus similar to the existing scram bus;

(b) connect the two scram buses to the scram solenoid valves in a manner which ensures that either bus is capable of deenergizing the solenoid valves regardless of whether the other bus is operable or failed;

(c) disconnect one of the nuclear instrumentation logic networks from the existing scram bus and connect it to the additional similar scram bus;

(d) transfer either the high pressurizer pressure or the high primary pressure trip relay contacts from the existing scram bus to the additional similar scram bus;

(e) install additional independent high primary temperature, low primary pressure, and low boiler feedwater pressure instrumentation for use with the additional similar scram bus;

(f) transfer one flux/flow computer from the existing scram bus to the additional similar scram bus and install additional flux averaging amplifiers, one for use with the additional similar scram bus and one for use with the automatic reactor control system;

(g) modify the existing scram bus and design the additional similar scram bus (or design their connections to the scram solenoid valves) in a manner that permits testing individually the two scram buses during reactor power operation; and

(h) provide redundant initiating devices in the criteria for protecting against the effects of a steam line break and for isolating containment.

If we can provide additional guidance with regard to your preparation of these required submittals, please do not hesitate to call (Mr. Roger Woodruff is the Regulatory Project Manager).

You should appreciate that, in addition, there are other technical issues related to your pending application for a full term operating license for Indian Point Unit No. 1 now under review by the AEC staff. In this connection, we are concerned with the frequent delays that have been experienced in your responses to our requests for additional information on these issues. We believe that this matter warrants prompt management attention and request that you take the necessary steps to remedy this situation.

Sincerely,

A. GIAMBUSSO,
Deputy Director for Reactor Projects,
Directorate of Licensing.

Enclosure:

Safety Evaluation by the Directorate of Licensing Consolidated Edison Company of New York, Inc.—Indian Point 1: Docket No. 50-3

Background

As part of our review of Consolidated Edison Company's application for a full-time operating license for Unit No. 1 at the Indian Point Nuclear Generating Station, we have completed a review to determine the need, if any, for backfit of the plant protection system. Car position and the reasons for it are discussed in the following sections.

Description of the Plant Protection System

Principal features of the Unit No. 1 plant protection system are a scram bus, two logic networks, and circuits for isolating containment and for isolating steam lines.

When values of important plant parameters exceed their limiting safety system settings, relays in the scram bus automatically cause insertion of the control reds by deenergizing a solenoid valve in the hydraulic system for each control red. Principal relays are controlled by instrumentation which monitors reactor coolant temperature and pressure, boiler feedwater pressure, and the ratio of neutron flux to reactor coolant flow.

Neutron flux sensors monitor reactor power and provide input signals to the logic networks. These networks supply electrical power to the scram bus and are intended to interrupt electrical power to the scram bus if reactor power exceeds the limiting safety system setting.

Multiple input signals to the logic networks are provided by each of 18 neutron flux sensors. The signals are connected in sets of three to each of nine "two-out-of-three" modules in each network. If any two signals to a "two-out-of-three" module exceeds the limiting safety system setting, the module initiates a scram signal. The outputs of the nine "two-out-of-three" modules are connected to a "one-out-of-nine" module. On receipt of a scram signal, a solid state switch in this module "opens" interrupting power to two trip actuator amplifiers. A solid state switch in each of the parallel trip actuator amplifiers then "opens", disconnecting power from the scram bus. Because the two logic networks are connected to the scram bus in parallel, both logic networks must function to deenergize the scram bus.

Three of the 18 neutron flux sensors also provide signals to a flux averaging amplifier. The output from this amplifier and signals from reactor coolant flow sensors are connected to two computers which calculate the ratio of flux to flow. If the value of the ratio is excessive, each computor demands that each of two pairs of parallel relay contacts opens in the scram bus. Other pairs of parallel relay contacts in the scram bus are opened by sensors monitoring boiler feedwater pressure, reactor cooling system pressure, and reactor cooling system temperature.

To prevent overcooling of the reactor core in the event of a steam-line-break accident and to limit the reactivity and power transients which would result, circuitry is provided to close the steam stop valve in the affected loop if a reactor scram occurs in coincidence with low steam pressure. A pressure drop across the steam stop valve in coincidence with scram causes closure of the stop valves in the associated reactor cooling loop. These functions are initiated by a single switch or relay.

Containment is partially or completely isolated automatically in the event of high containment pressure, high radiation, and very low pressurizer level. For each case, isolation is initiated by a single relay.

Safety Evaluation

IEEE 279 establishes several criteria for the design of plant protection systems. However, the fundamental aspects of IEEE 279 is the single failure criterion which is intended to assure that the protection system will function in the event that any single failure or fault exists somewhere in the protection system. Our safety evaluation is based on this fundamental aspect of IEEE 279.

With regard to the scram bus, application of stray power to it can disable all relays and solid state switches from the point of application of stray power back to, and including, the trip actuator amplifiers, thus, defeating the scram functions associated with the relays and solid state switches. Other kinds of single failures, e.g., failed components, short circuits, open circuits, and grounds, can also disable relays in the scram bus and cause less of specific scram functions.

In the logic networks, random failure of any one of six solid state switches in the "closed position" will disable the high neutron flux scrams. In the circuits for isolating containment and for isolating steam lines, single failures of initiating mechanisms would cause loss of protective action.

Although we have not reviewed all portions of the plant protection system, it is clear that single failures can be expected to inhibit the plant protection system and that backfitting the plant protection system will provide substantial, additional protection which is required for the health and safety of the public.

Backfit and Information Requirements

Pursuant to 10 CFR Part 50, Section 50.109, we are requiring that Consolidated Edison submit, within six months from the date of this evaluation, proposed modifications to the plant protection system which will provide significantly greater assurance that the plant protection system will perform its intended function should any of the following kinds of single failures occur: failed components, short circuits, open circuits, grounds, and application of any available ac or dc potential. The proposed modifications must include capability at power for conducting surveillance on the plant protection system to assure that it is capable of performing its intended function. Further, we are requiring that Consolidated Edison complete approved modifications of the plant protection system within one and one-half years from the date of this evaluation.

R. W. WOODRUFF,
Operating Reactors Branch No. 1,
Directorate of Licensing.

ROBERT J. SCHEMEL,
Chief, Operating Reactors Branch No. 1,
Directorate of Licensing.

APPENDIX 14

CORRESPONDENCE: TVA/NRC

LETTER DATED SEPTEMBER 2, 1975, TVA TO NRC, RESPONDING TO VIOLATION LETTER OF JULY 28, 1975

TENNESSEE VALLEY AUTHORITY,
Chattanooga, Tenn., September 2, 1975.

MR. NORMAN C. MOSELEY,
Director, Regional Office,
U.S. Nuclear Regulatory Commission, Atlanta, Ga.

DEAR MR. MOSELEY: We have reviewed the material provided with your letter to J. E. Watson dated July 28, 1975, (50-259/75-1, 50-260/75-1), related to the Browns Ferry fire. In accordance with your request, we have prepared responses to the alleged infractions in the Notice of Violation and items mentioned in Areas of Concern, Appendix A and Appendix B respectively, to your July 28 letter. In addition, we have prepared responses related to the Conclusions and Summary of Findings of Facts included in your Regulatory Investigation Report. The TVA responses are enclosed. With reference to Section 2.201, Items (1), (2), and (3), in the fourth paragraph of your letter, you will note that except in the cases of those allegations TVA has denied, the additional administrative procedures have already been placed into effect.

In addition to revisions in administrative controls, TVA has, as you know, also committed to numerous revisions to the Browns Ferry plant design to minimize the likelihood of a similar fire. These are discussed in great detail in TVA's "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" dated April 13, 1975. You have been provided copies of all such documents and correspondence and we assume I & E is familiar with the status of the plant.

Much of the controversy among investigators of the incident hinges on the timing of the use of water as an extinguishing agent on the burning electrical cables. The plant operators were concerned that additional components would be rendered inoperable if their cables were shorted out by spraying water. TVA has no general policy prohibiting the use of water on electrical fires, and we believe that our proposed design changes to the plant will enable water to be used in the future to extinguish cable tray fires without compromising plant safety.

The Browns Ferry Nuclear Plant is currently shut down with both reactor cores unloaded. TVA is pursuing as diligently as possible all actions necessary to put the plant back into operation as soon as possible, and this activity is requiring major effort by TVA management and staff. Because of these conditions, and because we believe the seriousness of the alleged infractions warranted more time for a thorough and measured response, we asked for additional time beyond September 2 (which you did not grant) to allow us to respond more fully. We hope that any future correspondence on this matter will consider the status of the plant and the overall recovery plan.

We believe our overall corrective action program has reduced to an extremely low level the probability of the occurrence of another fire at Browns Ferry, and has reduced to even lower levels the probability that a fire would result in a significant consequence.

Very truly yours,

J. E. GILLELAND,
Assistant Manager of Power.

Enclosure.

ENCLOSURE: NRC JULY 28, 1975, DOCKET NOS. 50-259, 50-260

NOTICE OF VIOLATION

1. FAILURE TO COMPLY WITH 10 CFR 50.59

Items appearing to be in noncompliance with 10 CFR 50.59, "Changes, Tests and Experiments," as indicated below:

a. 10 CFR 50.59, requires, in part, that records be maintained of changes to the facility to the extent that such changes constitute changes to the facility as described in the Safety Analysis Report. It further requires that these records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. The Browns Ferry FSAR Section 5.3.3.5 specifies, in part, that all electrical penetrations are sealed with sealant around conductors.

Contrary to this requirement, a safety evaluation was not made of the "change to the facility as described in the Safety Analysis Report" which was constituted by operation of the reactor with containment penetrations unsealed while concurrently sealing and testing the penetrations.

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

TVA Response

The Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR) was written to describe a completed 3 unit plant. To varying degrees, the FSAR describes most all plant features regardless of their nuclear safety significance.

Before issuance of the operating license for each unit, the NRC (then Regulatory Staff of the AEC) made findings that, among other things, the facility construction was substantially complete in conformity with the construction permit and the rules and regulations of the Commission. The Browns Ferry units were licensed and permitted to operate one unit at a time while construction continued on the remaining units. This was not considered to be a violation of the operating license or a violation of 10 CFR 50.59 since it was the basis upon which the licenses were issued.

TVA has made written safety evaluations in accordance with 10 CFR 50.59 before making changes to completed portions of the facility but has not been in the practice of making safety evaluations for work on components where construction work had never been completed except where a safety-related interface was recognized to exist. The section of the FSAR referenced in the allegation is a description of the secondary containment system describing not only the seals around electrical cables but all other building seals which contribute to the integrity of the secondary containment. Unsealing penetrations to add additional cable and resealing them upon completion was not considered as a change to the facility design nor was it recognized to have safety significance as long as the limiting conditions for operation of the secondary containment described in the technical specifications were maintained.

To reduce the potential for this type occurrence in the future, TVA has made procedural changes to require additional review before activities are authorized. The changes in procedures which have been presented to NRC are discussed in "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)," a copy of which has been provided you.

2. FAILURE TO COMPLY WITH TECHNICAL SPECIFICATIONS

Items appearing to be in noncompliance with the facility Technical Specifications, as indicated below:

a. The Technical Specifications, Sections 6.3.A and 6.3.B state, in part:

"A. Detailed written procedures, including applicable check-off lists covering items listed below shall be prepared, approved and adhered to * * *.

"4. Emergency conditions involving potential or actual release of radioactivity * * *.

"5. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.

"6. Surveillance and testing requirements * * *.

"B. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation * * * Such changes shall be documented and subsequently reviewed by PORC and approved by the plant superintendent."

(1) Contrary to these requirements, the leak testing, sealing and inspection of the penetrations were being accomplished; but detailed written procedures approved by the plant superintendent and reviewed by PORC had not been developed for the control of this work.

TVA Response

It was not recognized that this work on the penetrations was of such a nature as to have an effect on the safety of the reactor, on surveillance or testing requirements, or had the potential for release of radioactivity; therefore, leak testing, sealing, and inspection of the penetrations were not accomplished by detailed, approved, written procedures as would be required for safety-related items. Procedural changes have been made which require greater in-depth review of all significant activities prior to their authorization. In addition, procedural changes have been made which require review of plant activities by the plant Quality Assurance Supervisor to verify that the activities are supported by approved instructions necessary for their performance.

(2) Contrary to these requirements, persons discovering the fires on March 20 and March 22, 1975, did not adhere to the provisions of the Emergency Procedure in that they did not initiate the fire alarm.

TVA Response

The construction workers first attempted to extinguish the fire, whereas the Browns Ferry Nuclear Plant Emergency Procedure specifies that the fire alarm be sounded first. The officer reporting the fire telephoned the shift engineer's office rather than calling either of the numbers listed in the procedure. To correct this condition, greater emphasis on emergency procedures has been and is being stressed as part of the Browns Ferry Nuclear Plant employee orientation program and fire drills accompanied by instruction over the public address system are being utilized as a means of training all employees in the proper action to take in case of fire.

(3) Contrary to these requirements, the Browns Ferry Emergency Procedure was not adhered to in that the Shift Engineer did not delegate onscene responsibility for fire fighting to an assistant shift engineer when he departed the fire area.

TVA Response

TVA denies this allegation and believes that the Browns Ferry Emergency Procedure was followed. The shift engineer and assistant shift engineer both reported to the scene of the fire. The shift engineer was recalled to the control room by the reactor operator leaving the assistant shift engineer in charge of fire-fighting activities. The assistance shift engineer was automatically in charge when the shift engineer was called away, the assistant shift engineer knew he was in charge, and his actions support that he assumed this responsibility. The NRC investigation report substantiates that the assistant shift engineer coordinated fire-fighting efforts, evacuated people from the area, made reports to the shift engineer, and placed the call for offsite assistance.

(4) Contrary to these requirements and the requirements of Browns Ferry Standard Practices Manual which specify, in part, in Standard Practice BFS3 that:

"Plant fire protection systems shall be fully operational at all times. Removal of a plant fire protection system from service for any reason other

than as required in a test procedure requires the approval of the plant superintendent. Removal of a system from service for more than seven days requires a review of PORC."

The fire protection system for the cable spreading room was not fully operational in that metal plates had been installed under the glass in the manual stations during the construction of the plant and had not been removed. The approval of the installation of the plates had not been documented prior to or subsequent to the issuance of the operating license and the installation had not been reviewed by PORC. Additionally, the CO2 manual-automatic initiation system had been electrically disabled by the construction workers without documented approval of the Plant Superintendent.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

TVA denies this allegation; the CO_2 system was not removed from service. The presence of the metal plates under the breakout glass used in the mechanical valve method did not prevent manual electric actuation nor slow application of the CO_2. The CO_2 system in the spreading room can be actuated manual electrically or manual mechanically. The system does not have automatic actuation. Of the two means available to actuate manually the system, the operator used the manual electrical method, which is the normal method and the faster method. The method used (in accordance with established instructions) was significantly faster than the mechanical valve method. As a safety precaution, the metal plates had been installed to prevent inadvertent operation of the CO_2 system in the spreading room while workers were present. The condition of the CO_2 system was given consideration and determined not be contrary to the standard practice requirements in that the CO_2 flooding system was operational and not removed from service. In addition, written plant instructions were in effect describing the condition of the system and specifying its mode of operability.

3. FAILURE TO COMPLY WITH APPENDIX B TO 10 CFR 50

Items appearing to be in noncompliance with Appendix B to 10 CFR 50, "Quality Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants," as indicated below:

a. Criterion XVI of Appendix B to 10 CFR 50 and the related commitments in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," Section D.4.2.4.7 specifies, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected; that measures assure that causes of conditions be determined and action taken to preclude repetition; and that the corrective actions are documented and reported to appropriate levels of management.

Contrary to these requirements, during the penetration sealing operations, the conditions adverse to quality were not promptly identified and corrected; the causes of conditions were not determined and actions taken to preclude repetition; and the required documentation was not supplied in the two instances that fires were reported to management.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The minor fire that occurred in the sealing material on March 20, 1975, had been discussed in the plant management staff meeting on March 21 and was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred. Plant administrative instructions for reporting conditions adverse to quality have since been revised and are also included as a part of the general employee training program.

b. Criterion X of Appendix B to 10 CFR 50, requires, in part, that a program for inspection of activities affecting quality be established to verify conformance with documented instructions, procedures, and drawings; and that persons assigned the responsibilities for such inspections shall be independent of individuals directly responsible for work performance. Related commitments are spelled out in the FSAR, Appendix D.4., Sections D.4.2.3.1 and D.4.2.1.1., respectively.

Contrary to these requirements, inspections of the sealing of cable penetrations were not conducted so as to assure conformance with drawings; and in-

spectors were involved in the work activities for which they had inspection responsibilities.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The response to item 2.a(1) is also applicable to this item. Since the fire, procedures have been issued applicable to all penetrations under repair or construction. These procedures provide for instruction of craft personnel engaged in penetration sealing and flameproofing and provide for independent inspection by engineers. The inspectors are prohibited from engaging in the work.

c. Criterion XVIII of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," specifies, in part, that a comprehensive system of planned audits be carried out to assure compliance with all aspects of the quality assurance program.

Contrary to these requirements, a review of the records of the audits conducted at Browns Ferry and discussions with responsible individuals indicated that no audits had been conducted of the penetration installation.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

Audits are not required for non-safety-related items, therefore, audits were not conducted of the records pertinent to penetration installation because their safety significance had not been recognized. Since the fire, a detailed audit has been made of all penetrations and work of this nature has been added to the quality assurance audit program.

d. Criterion III of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.2, "TVA Quality Assurance Plan for Design and Construction," Section D.2.4.3.4., specifies, in part, that certain basic design drawings, such as single line diagrams, are reviewed to determine that they meet the design bases, design criteria and other design input requirements.

The FSAR, Amendment 25, "Response to AEC Question 7.5," states, in part, that cables for the Engineered Safeguards Systems are separated into two redundant divisions (Division I or Division II) such that no single credible event could damage the cables of redundant counterparts. This section further states that power cables from the 4160-Volt Shutdown Boards are installed in separate conduits. It further states that the electric circuits of one of the two loops of the Core Spray System including the pump motors and electrically operated valve, are in Division I; and the circuits of the other loop are in Division II. Additionally, it states that the electric circuits associated with pumps A and C, and their valves, of the LPCI sytsem are in Division I; and the electrical circuits of pumps B and D, and their valves are in Division II.

(1) Contrary to this requirement, the power cable supplying 480 Volt Shutdown Board 1B from 4KV Shutdown Board C (Division II) is routed in the same tray as the power cable supplying 480 Volt Shutdown Board 2A from 4KV Shutdown Board B (Division I).

TVA Response

The routing indicated in the finding was not a violation of the separations criteria and the electrical design was reviewed and approved by the NRC. This finding is an oversimplification of a highly technical point. The original design did not assign divisional classification to 4-kV cables between the 4-kV shutdown boards and the 4-kV/480-volt transformers. These cables were assigned an IE designation. Separation requirements for IE cables are that the normal and alternate feeders for the *same* board will not be run in the same tray. The design was analyzed to and met the single failure criterion.

(2) Contrary to this requirements, RHR Pump 1C and Core Spray Pump 1C are supplied from 4KV Shutdown Board B, and their associated valves are supplied from 4KV Shutdown Board C. Shutdown Board B is in Division I and Shutdown Board C is in Division II.

This infraction had the potential for causing or contributing to an occurrence related to safety.

TVA Response

The IE report contains a mistake and therefore erroneously concludes that separations criteria were violated. Core Spray Pump 1C is supplied from 4 kV

Shutdown Board B, as stated, but power for the valves associated with this core spray pump is supplied from 4-kV Shutdown Board A, not Board C. Power for the pump and its associated valves is taken from Division I.

The RHR pumps have three modes of operation, namely torus cooling, shutdown cooling, and low-pressure coolant injection (LPCI). RHR Pump 1C is supplied from 4-kV Shutdown Board B in Division I as stated. In the first two modes of operation, power for the associated valves is supplied from 4-kV Shutdown Board A, which also is in Division I. In the LPCI mode, the injection valve (FCV 74-73) associated with this pump receives power from 480-volt Reactor MOV Board 1C which receives its normal power from 4-kV Shutdown Board C through 480-volt Shutdown Board 1B. This 480-volt shutdown board supplies Division II loads. Reactor MOV Board 1C has an automatic transfer capability upon loss of normal power, and would transfer from 480-volt Shutdown Board 1B to 480-volt Shutdown Board 1A which is supplied from 4-kV Board A. This 480-volt shutdown board supplies Division I loads. The automatic transfer feature of 480-volt Reactor MOV Board 1C permits the power supplies to be arranged as described without violation of separations criteria. This design feature was received and approved by NRC.

Appendix B—Areas of Concern

1. Construction personnel were involved in work using open flame but had not been given fire fighting training nor did they have personnel with this training assigned to their area.

TVA Response

Fire fighting in an operating unit is provided by trained plant operations personnel. Construction personnel are not required to fight fires in an operating unit and therefore do not receive fire fighting training. Procedures have been changed such that when working in an operating unit area, they do receive orientation in emergency procedures, including reporting of fires. Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for airleaks. In addition, a procedure has been written establishing control over the use of cutting, welding, and open flames in operating plants. This procedure requires a physical survey of the areas where the work is to be performed and the establishment of safeguards including additional fire fighting equipment and a fire watch prior to any work involving open flame or welding.

2. Construction personnel working in the plant were not familiar with the plant emergency procedures, and some were not familiar with the plant layout.

TVA Response

The response to alleged violation 2.a.(2) is also applicable to this concern.

3. Operations personnel on shift were not fully aware of the ongoing construction activities.

TVA Response

The shift engineer on duty was aware of the ongoing construction activities through the information contained in the workplans. A copy of each workplan is held by the shift engineer, and all work on transferred equipment is scheduled with him. Because the area where the fire was initiated is normally kept locked, the shift engineer's approval was required prerequisite to entry. Since the fire, procedural changes have been made requiring workplans to contain a more detailed description of the work to be performed. In addition, all workplans now receive supervisory and quality assurance review, are listed on a daily activity sheet, and their performance requires authorization of the plant superintendent.

4. Plant operations personnel and Public Safety Services personnel at the plant were not familiar with the reactor building fire extinguishing equipment in that they did not know that the nozzle on the hose was specifically designed for fighting electrical fires.

TVA Response

Plant operations personnel who have responsibility for fire fighting are instructed in the use of fire extinguishing equipment. Fog-type nozzles are utilized in our fire training classes, and an individual would only have to look at the nozzle to see the type it was. The only statement we can find to support the conclusion that personnel were not familiar with the fire hose nozzle is in the I&E

report where an operator in his statement said it worked more like a shower head than a fog nozzle.

5. Emergency breathing apparatus appears not to have been properly maintained.

TVA Response

TVA contends the emergency breathing apparatus was well maintained. Plant procedures for the care of this equipment were in effect and were being used. Some air cylinders which were in the shop for reconditioning and were specifically marked were inadvertently brought to the scene of the fire, but they were not used and did not endanger personnel safety.

6. Certain of the operations personnel as well as the construction personnel were not familiar with use of the breathing apparatus.

TVA Response

Plant operating personnel are responsible for fire fighting in an operating unit. Construction personnel are not normally trained in the use of air-breathing equipment. This is a part of the training provided operations personnel and the effectiveness of this training can be adjudged by the total reliance on air-breathing equipment by employees engaged in fire fighting and other activities associated with safe reactor shutdown without fatality or significant personnel injury.

7. There was apparently no attempt by CECC management to obtain expert advise on methods for fighting the fire.

TVA Response

The CECC is not designated the responsibility for advising on fire fighting in the Radiological Emergency Plan. The DECC had as part of their staff considerable expertise for judging the probability of short circuiting vital equipment by using water. The use of water was discussed several times throughout the fire with the Division Emergency Control Center (DECC). The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors and to minimize personnel injury. The report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam, or straight) because of the possibility of shorting circuits and further degradation of the plant to a condition that would have been more difficult to control. Reactor safety concerns under the circumstances took precedence over extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as a correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached.

8. A design review subsequent to the fire revealed two cable trays with cables installed in excess of the design criteria. The practice of the construction forces of abandoning cable in-place cor'd result in additional cable trays being overfilled.

TVA Response

The cases indicated do not constitute violations of the cable tray fill criteria. Question 7.5 of May 22, 1971, requested that TVA provide the criteria and their bases which establish the minimum requirements for preserving the independence of redundant reactor protection systems, engineered safety feature systems, and Class 1E Electrical Systems through physical arrangement and separation, and assure minimum availability during any design basis event. In answering this question, we included a statement that cable trays carrying control and signal cables are loaded to a maximum of 60 percent of the cross-sectional area of the tray. This answer applied to the safety related cables referenced in the question.

The 60 percent guideline figure, which is the ratio of the cross-sectional area of the cable to the cross-sectional area of the tray, [illegible]s obtained from past experience wherein it had been determined that a cable [illegible]ray would be approximately level full at this point. Physical space for cables was the primary consideration because the weight limitation and electrical current considerations contained ample margin for filling to the worst case. If the cable tray became full before the 60 percent guideline was reached because of cable configuration in the tray no additional cables were routed in that tray. The same general guideline for non-safety-related cables was also followed although it is not specifically addressed in the FSAR.

Because of certain factors such as number of various sizes of cable or special installations in trays, it was possible that some trays would not be full when the 60 percent value was reached. In some cases, the 60 percent guideline was exceeded up to the point of the tray being level full by installing cables in these trays with special care. Although two cable trays are referenced, one is actually a continuation of the other.

9. The Browns Ferry Emergency Procedures and Standard Practice Manual contain errors and inconsistencies relating to emergency response. The Standard Practice Manual, BFA-34, permits the DPP Coordinator to review work plans for safety significance and does not require further review of his evaluations. This could result in a Technical Specification violation.

TVA Response

This item appears to contain a clerical error in report preparation in that the statements do not seem to be related. However, we believe one concern intended is that plant procedures contained two different phone numbers for use in reporting a fire. One phone number initiates the fire alarm and is a standard number used in all TVA operating plant phone systems. The other number on the construction phone system rings the fire marshall responsible for fire fighting on units under construction and before the operating license is received. Dialing either on the appropriate telephone system would result in prompt reporting of a fire. However, instructions existing within the plant operating area have been revised to delete the phone number of the construction fire marshall. The concern related to work plan review is addressed in the TVA response to alleged infraction 2.a.(1).

10. There was apparently no attempt by the CECC director to direct the operations of the center. The plant superintendent was involved in extensive communications with the center rather than in directing the plant fire fighting and recovery operations. As a consequence, the efforts at both locations were not coordinated, management personnel were not provided with accurate information and their actions were relatively ineffective.

TVA Response

TVA denies this allegation.

The TVA Radiological Emergency Plan (REP) sets forth the policies; purposes; delegations; standards; guidelines; and, where feasible, specific instructions necessary for TVA to discharge its responsibilities during a radiological emergency in order to comply with pertinent directives applicable to the protection of TVA personnel, plants, and properties and the health and welfare of the general public.

The general delegation of responsibility among the various emergency staffs as outlined in the Browns Ferry Nuclear Plant Radiological Emergency Plan are as follows:

A. Central Emergency Control Center (CECC)

Ensure that the various staff members man their positions.

Notify and inform other Government agencies.

Establish communications with Division Emergency Director and Environs Emergency Director.

Protect the public.

Gather offsite information.

Review with Division Emergency Director measures taken to control emergency.

B. Division of Power Production Emergency Center (DECC)

Receive reports from Site Emergency Director.

Evaluate and make recommendations.

Ensure that support efforts are adequate.

C. Site Emergency Staff

Assume responsibility for emergency.

During the Browns Ferry fire, these responsibilities were carried out as designated. There was, according to the logs of various staffs, information and direction given between the individuals involved.

Although the alternate CECC director did not realize immediately on his arrival that he was acting as director at the CECC, his logs reveal that he, upon notification (1604 EDT), contacted the Environ Director and directed him to

notify the Alabama Department of Public Health. This delay in recognition did not slow down the Radiological Emergency Plan from being implemented.

Upon his arrival at the CECC (1625 EDT), the alternate CECC director was briefed by the director, Division Emergency Center, on plant status. The public information officer was briefed. NRC was called at 1703 EDT. Personal contact with the Alabama Department of Public Health was made at 1712 EDT. In brief, coordination was evident and all divisions of responsibility were being carried out. This is evidence of coordination of positive leadership.

The focus of communications with other agencies centered on plant and environmental conditions. We do not believe this can be considered a misdirection since one directly affects the other. Since data provided indicated no radioactive release, the various agencies requested information on plant status and used this information to assess the potential for environmental release.

The status of the fire at Browns Ferry was discussed numerous times according to the Division Emergency Center log and the plant log. The information was not always timely because the fire was reported out and then would flare up again. This fact may have caused some confusion, but communications on reporting the conditions of the fire were successful even though other plant problems took precedence at times.

As pointed out in preface statements, fire fighting efforts were the responsibility of the Site Emergency Director with advice from the division emergency center. At no time was control of the fire beyond the scope of the resources already under the direction of the plant superintendent (site emergency director). There is no basis for stating that CECC guidance was inadequate.

The plant superintendent could not have personally directed all fire fighting activities nor does good management technique suggest that he should. Had he attempted to do so he would not have been able to carry out other specified responsibilities which have a much broader impact on reactor safety and the health and safety of the public. The superintendent was not engaged in *extensive* communication with the center, and this communication did not impair the superintendent's performance of his duties and responsibilities.

11. Although plant procedures specify that the Unit operator has the authority to shut down the reactor in the event of an emergency, his responsibility to effect a shutdown promptly has not been defined by management.

TVA Response

TVA denies this allegation.

Plant instructions specify that the reactor operator is responsible for the operation of his unit and give him authority to shut it down when he deems its safety is in jeopardy. Further, his job description specifies that, in emergencies where there is not time to get advice from his supervisor, he may operate on his own initiative to correct existing conditions or to save life or property. It is impractical and probably impossible to define in detail all conditions for which prompt shutdown is desirable.

CONCLUSIONS

A. General

1. The radiological impact on the public, plant personnel or the environment resulting from this fire was no more significant than from the routine shutdown of the reactors.

TVA Response

This finding is correct. Priority efforts at the plant were directed towards the primary objectives of ensuring safety and minimizing personnel injury and these objectives were achieved.

2. Both Unit 1 and Unit 2 cores remained adequately cooled throughout this occurrence. Although some systems were rendered inoperable, plant operators were successful in bringing into operation alternate means to cool the reactors.

TVA Response

This finding is correct. Priority efforts at the plant were directed towards ensuring safety and minimizing personnel injury.

3. Some minor injuries occurred as a result of the fire.

TVA Response

A few Browns Ferry employees were treated for smoke inhalation. No member of the public was injured in any way.

B. Direct Cause of the Fire, its severity or its consequences

1. The following factors contributed directly to the cause or severity of the fire:

(a) Failure to evaluate the hazards involved in the sealing operation and to prepare and implement controlling procedures.

TVA Response

Prior to operation of Browns Ferry unit 1, TVA engineers participated in and reviewed reports of fire tests made on penetrations and cables similar to those which were to be used at Browns Ferry. The test results indicated that fire propagation through a completed penetration was an unlikely event. On the basis of these tests, a penetration and cable fire was not considered to be a high degree risk; therefore, special design control and installation procedures were not considered necessary.

Since the March 22, 1975, fire, TVA management is aware that cable and penetration fires are a potentially severe hazard. Administrative control and inspection have been upgraded in consideration of the hazard.

(b) Failure of workers to report numerous small fires experienced previously during the sealing operations, and the failure of supervisory personnel to recognize the significance of those fires which were reported, and to take appropriate corrective actions.

TVA Response

In the future the procedures to be followed in fire reporting will be stressed to all employees at the plant as a part of the overall training program. A minor fire that occurred in the sealing material on March 20, 1975, has been discussed in a plant management staff meeting on March 21 and was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred.

(c) Use of an open flame without fire precautions specific to this activity.

TVA Response

Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for air leaks. In addition, a procedure has been written establishing the control of the use of cutting, welding, and open flames in operating plants.

(d) Ineffectual leadership by TVA in fire fighting activities.

TVA Response

Since the fire additional plant supervisory personnel have attended the fire brigade leader training course so that now all Browns Ferry Shift Engineers and Assistant Shift Engineers are certified as having participated in this training.

(e) Inadequate training of TVA personnel in fire fighting procedures and equipment.

TVA Response

TVA believes there was adequate training in fire fighting procedures and equipment. Apparently training records at Browns Ferry were not reviewed by the NRC even though the records were made available to the investigators. Extensive training in fire fighting techniques had been conducted before the fire and the records so show. In addition, a special briefing on fire fighting has been conducted for over fifty Browns Ferry operations personnel since the fire.

(f) Delay in application of water in fighting the fire.

TVA Response

The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors thereby insuring public health and safety and to minimize personnel injury. The NRC Office of Inspection and Enforcement report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam, or straight) because of the possibility of short circuits and further degradation of the plant to a condition that would have been more difficult to control. Reactor safety concerns under the circumstances took precedence over extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as the correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached at the plant.

(g) Difficulties encountered in use of self-contained breathing apparatus, caused by inadequate training of personnel in its use, inadequate maintenance and inability to recharge air bottles fully.

TVA Response

The types of self-contained breathing apparatus provided at Browns Ferry are well maintained, are fully adequate, and are in common use in fire fighting. A special training course has been conducted for plant employees in the use of the equipment. Some air cylinders brought to the scene of the fire had been specifically marked as being in the shop for reconditioning. Additional recharging equipment will be installed at the plant to provide additional capability onsite to recharge a large number of tanks fully and more rapidly.

(b) Inaccessibility of the initial ignition location due to the position of the penetration and testing of seals prior to fireproofing.

TVA Response

The location of the fire barrier in the penetration has no bearing on the function of the fire barrier. While technically it would be a deviation from the design drawing, it would not be a reportable design deficiency.

(c) Lack of familiarity with the performance characteristics of the fire nozzle provided for such fires and ultimately used successfully to fight the fire.

TVA Response

Plant Operations personnel who have responsibility for fire fighting are instructed in the use of fire extinguishing equipment. The only evidence that might support this conclusion is that one operator in his statement said it worked more like a shower head than a fire nozzle. We do not understand the basis for this conclusion. Fog-type nozzles are utilized in our fire training classes and an individual would only have to look at the nozzle to see the type it was.

2. The following factors contributed to difficulties experienced during the post-shutdown cooling of the reactors:

(a) For Unit 1, fire rendered inoperable for a significant period of time portions of the control and power circuits for components used in the normal cool-down of the reactor and in emergency core cooling systems.

TVA Response

The Emergency Core Cooling System (ECCS) is designed to provide adequate core cooling in the unlikely event of a loss of normal core cooling water including a postulated break in the largest reactor cooling water pipe during full power operation. This accident, of course, did not take place. While the availability on unit 1 of emergency core cooling water throughout the event would have greatly facilitated unit 1 core cooling, the reactor was safely shut down using other cooling water sources in conjunction with the operation of the reactor depressurization features of the ECCS. Adequate ECCS for unit 2 was available.

(b) Lack of initiative on the part of onsite supervisory personnel to coordinate shutdown activities until after the first manual initiation of depressurization of Unit 1.

TVA Response

An objective review of the detailed statements included in the NRC I&E report will substantiate that supervisory management personnel were onsite and did exercise the initiative to coordinate the activities pertaining to reactor shutdown and restoration; that the plant superintendent was in the control room and states that the decision to blowdown (reduce reactor pressure) was his; and that once the fire was extinguished an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to vital equipment.

(c) Lack of knowledge of location and severity of the fire on the part of control room personnel and delay in rapid shutdown of the reactor when there were anomalous instrumentation indications.

TVA Response

As noted in Conclusions A1 and A2 both units 1 and 2 were safely shut down and cooled down. See TVA response to Area of Concern No. 11.

(d) Reduction of the options for cooling available during the period of repressurization of the Unit 1 reactor cooling system.

TVA Response

Overpressure protection was available at all times for both reactors. On Unit 1 the capability to blowdown the reactor to low pressure was lost for a period of time. The NRC is reviewing a TVA-proposed design change to eliminate this problem in the future.

(e) Failure to establish priorities for restoration of equipment. This resulted in uncoordinated and, in some cases, counterproductive or duplicative attempts to restore key equipment to service.

TVA Response

An objective review of the detailed statements included in the I&E report will substantiate that supervisory management personnel were onsite and did coordinate the activities pertaining to reactor restoration. Once the fire was extinguished, an orderly approach to verifying actual conditions was undertaken and temporary power was reestablished to vital equipment.

As wires were disconnected or contacts jumpered, the steps were written down and later put in a log as required by standard plant practices. In addition, the Plant Operations Review Committee met at 8:45 the night of the fire to discuss plant conditions and restoration activities to be taken during the night. The few examples cited of counterproductive or duplicative effort in the I&E transcripts are overwhelmed by statements to the contrary.

3. The following factors contributed to the adverse interaction between safety-related systems of Unit 1 and Unit 2:

(a) Two identified cases of failure to follow the TVA cable separation criteria described in the FSAR.

TVA Response

See TVA response to alleged infraction 3.c (1).

(b) Some Unit 1 and Unit 2 ECCS equipment is supplied from common 4 KV boards in accordance with the FSAR. The failure of any of these boards causes a loss of power to the connected equipment in both units.

TVA Response

It is important to note that there are two divisions of redundant emergency core cooling equipment for each reactor. Either division can provide adequate core cooling in case of a design basis accident. Failure of a single 4 KV board would affect, at most, equipment in one division on each unit.

This is a well known design feature of the Browns Ferry plant. It was thoroughly reviewed by the Atomic Energy Commission and was approved by the AEC as being completely acceptable. See TVA response to alleged infraction 3.d (2).

C. Other Important Factors Not Directly Contributing to the Cause of the Fire, its Severity or its Consequences

1. Rapid manual operation of the installed carbon dioxide system in the cable spreading room was precluded because metal plates installed behind break-out glass during construction had not been removed.

TVA Response

See TVA response to 2.a (4).

2. Operation of TVA's Central Emergency Control Center (CECC) was not well coordinated; information sometimes was not exchanged internally; CECC communications with personnel at the Browns Ferry site were not effective in keeping the CECC currently informed concerning the status of the plant or of recovery activities. Consequently, communications with other agencies led to misunderstanding of plant status by those agencies.

TVA Response

TVA believes that the CECC, Division of Power Production Emergency Center, the Site Emergency Staff, and the Environs Emergency Center fulfilled their responsibilities and that good coordination existed throughout the fire and recovery period. The information available to CECC was expeditiously communicated to outside agencies with the full knowledge that, in retrospect, some of the information could have been incomplete. There was no deliberate attempt by TVA to mislead outside agencies.

SUMMARY OF FINDINGS OF FACTS

A. Events Leading to the Fire and Fire Fighting Efforts

During the investigation of events leading to the fire and the fire fighting efforts, the following facts were disclosed:

1. The immediate cause of the fire was the ignition of polyurethane which is used at BFNP as a cable penetration sealing material. Construction workers

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were checking for leaks in a penetration connecting the cable spreading room with the reactor building. Personnel performing the checks utilized a candle flame to detect air flow from the cable spreading room to the reactor building. The candle flame ignited the polyurethane. (Details I, page 3)

TVA Response

Since the fire, TVA has issued written instructions prohibiting the use of any flame at Browns Ferry to check for air leaks. In addition, a procedure has been written establishing the control of the use of cutting, welding, and open flames in operating plants.

2. There was no approved written procedure provided to control the inspecting, sealing, and testing of cable penetrations. (Details I, page 11)

TVA Response

Since the fire, procedures have been issued to require surveillance by engineering personnel of all penetrations under repair or construction. These procedures also provide for instruction of craft personnel engaged in penetration sealing and flameproofing and provide for inspection by engineers.

3. Personnel responsible for inspecting the penetration seals were also involved in making the seal repairs. (Details I, page 11)

TVA Response

Procedures for installation and inspection of penetration seals have been issued that provide for installation by craft personnel and independent inspection by engineers. The inspectors are prohibited from engaging in the work.

4. Some penetration seals were not properly completed prior to operation of Units 1 and 2, nor were they properly maintained subsequent to operation. (Details I, page 9)

TVA Response

See TVA response to item 1.a.

5. The location of the fire barrier in the penetration did not conform to the approved design drawings. (Details I, page 9)

TVA Response

The location of the fire barrier in the penetration has no bearing on the function of the fire barrier. While technically it would be a deviation from the design drawing, it would not be a reportable design deficiency.

6. The electrical design specified the installation of cable in trays in excess of that permitted by the TVA design criteria in at least two cases. Other trays could contain excess cable in that cable was abandoned in place when designs were changed. (Details I, page 12)

TVA Response

A design guideline for cable tray fill based on 60% of the cross sectional area of a tray carrying control and signal cables is indicated in the FSAR. See TVA response to Area of Concern 8.

7. Although not required by license conditions, laboratory tests had been performed by TVA to verify the ability of the penetration seals to perform as a fire barrier. These tests, however, did not simulate the condition existing at the time of this fire. For example, the tests were not conducted with a pressure differential across the seal, the polyurethane was not exposed to the flame nor was a leakage path established through the seal. Additionally, the test did not include the sheet type polyurethane in use in the cable penetrations. The use of this sheet type foam had not been approved by the TVA design department. (Details I, page 10)

TVA Response

Flame tests were conducted before the operation of Browns Ferry to confirm the completed penetration seal design concept. The results of these tests indicated that a completed fire stop would be effective. Since the fire, TVA has constructed a facility to conduct testing under a differential pressure on a mockup of the penetration in which the fire occurred.

8. Construction personnel were utilizing an open flame in an area which is vulnerable to fire, but specific precautions and trained fire fighters were not present at the time. (Details I, page 12)

TVA Response

Written procedures have been established since the fire to control the use of open flames. In addition, instructions are being issued to require additional surveillance by fire fighting personnel at open penetrations in operating areas.

9. Previous fires in the polyurethane foam materials had not always been reported to the appropriate levels of management, and, on the occasions when reported, no action was taken to prevent recurrence. (Details I, page 11)

TVA Response

In the future the procedures to be followed in fire reporting will be stressed to all employees at the plant as a part of the general training program. A minor fire that occurred in the sealing material on March 20, 1975, had been discussed in a plant management staff meeting on March 21 and the minor fire was in the process of being followed up by supervisory personnel when the major fire of March 22 occurred.

10. The various procedures for responding to fires contained conflicting information. For example, the BFNP Emergency Procedure lists two different telephone numbers to be used in reporting a fire, one in a table of emergency numbers and the second in the test of the procedure. The appropriate number is the one in the test; dialing this number automatically sounds the fire alarm and rings the Unit 1 operator's telephone. (Details I, page 4)

TVA Response

At Browns Ferry two different phone numbers can be used to report a fire. One phone number initiates the fire alarm and is a standard number (299) used in all TVA operating plant phone systems. The other number on the construction phone system (235) rings the fire marshall responsible for fire fighting during construction. Dialing either on the appropriate telephone system would result in prompt reporting of a fire.

11. The BFNP Emergency Procedure was not followed by those involved when reporting the fire. The construction workers first attempted to extinguish the fire, whereas the procedure specifies that the fire alarm be sounded first. The guard reporting the fire telephoned the Shift Engineer's office rather than calling either of the numbers listed in the procedure. (Details I, page 4)

TVA Response

The procedures to be followed at the plant will be stressed as part of the general employee training program.

12. During construction, metal plates were installed under the breakout glass in the CO^2 system manual crank stations located at the entrances to the cable spreading room. These metal plates were not removed prior to plant operation, thus preventing manual initiation of the CO^2 system had it been necessary. The CO^2 system was successfully initiated automatically. (Details I, page 3)

TVA Response

See TVA response to alleged infraction 2.a (4).

13. TVA does not have outside agencies inspect their fire protection equipment and systems. Internal inspections of fire protection equipment had not revealed the presence of the metal plates. (Details I, page 6)

TVA Response

TVA had fire protection instructions in effect explaining fire protection system status including presence of the plates to the operators. The presence of the plates was known by TVA plant operations personnel. See response to Summary of Findings of Facts item A12.

14. Contrary to good safety practices, the plant procedures do not restrict the use of elevators during fires; both operations and construction personnel used the plant elevators while the fire was in progress. (Exhibit 1, Elect.-D, AUO-Q et al.)

TVA Response

We question the relevance to nuclear safety of this comment. Use of the elevators during the early stages of the fire enhanced rapid movement of personnel to perform operations related to safely shutting down the reactors and assisted in fighting the fire.

15. The practice at TVA's facilities has been that manual application of water on electrical fires be avoided for personal safety reasons. CO_2 and dry chemical equipment is provided for use by personnel fighting such fires. TVA's Fire Safety Manual permits the use of water on electrical fires and a nozzle designed for manual application of water on electrical fires had been provided at the hose station in the vicinity of the fire area. (Details I, page 13)

TVA Response

TVA has established no general policy prohibiting the manual use of water on electrical fires for personnel safety reasons. In this fire incident, the use of water was considered, but delayed due to reactor safety considerations. Written procedures have been revised and special training sessions held to emphasize that water is an acceptable extinguishing agent for electrical fires under appropriate circumstances.

16. The BFNP fire hoses and nozzles were not interchangeable with the Athens Fire Department equipment because different types of threads were used. (Details I, pages 8 and 13)

TVA Response

The fire hose threads at Browns Ferry were threads cut to a national standard. A particular nozzle from the Athens Fire Department would not mate with a Browns Ferry hose apparently because it was cut to another national standard. The TVA hose and nozzle were used to extinguish the fire. TVA will insure that fire hose threads at Browns Ferry are compatible with those of the Athens Fire Department.

17. Some TVA personnel experienced difficulty in using the self-contained breathing apparatus. (Details I, pages 6, 9 and 13)

TVA Response

See TVA response to Area of Concern 5.

18. There were no organized efforts to extinguish the fire in the reactor building for about three and one half hours. Sporadic individual efforts may have been made during this period. (Details I, page 7)

TVA Response

During this period, which was about two hours, the plant staff devoted its major efforts to the more important priority tasks of safely shutting down the reactors and fighting the fire in the cable spreading room after which fire fighting in the reactor building was resumed.

19. Although the responsibility for handling this occurrence rested with the Plant Superintendent, he was hesitant in authorizing the use of water on the fire; during discussions with his supervisor he requested permission to use water on the fire. (Details I, page 8 and Details IV, page 1)

TVA Response

The main objectives of TVA's efforts on March 22 were to ensure the safety of the reactors and to minimize personnel injury. The report fails to recognize these points in the repeated implications that water should have been used on the fire at an earlier time. The plant superintendent made the conscious decision not to use water (fog, foam, or straight) because of the possibility of short circuits and further degradation of the plant to a condition that would have been more difficult to control. Under the circumstances reactor safety concerns were a more important consideration than extinguishing a localized fire. Even in retrospect and in the light of nearly five months of engineering evaluation, this decision still stands as the correct decision. Of course, water was used to extinguish the fire after a more stable condition had been reached.

20. Only minor injuries to personnel were sustained during the fire. (Details I, page 18)

TVA Response

This finding is correct. See response to Conclusion A3.

21. The fire fighting and the shutdown of Units 1 and 2 were accomplished without excessive exposure of personnel to radiation and without the release of radioactive effluents from the plant in excess of license limits. (Details I, pages 13, 18)

TVA Response

This finding is correct. See response to Conclusion A1.

B. Operation Events and Problems Experienced During Fire and Until Shutdown Cooling Established

During investigation of operational events and problems experienced during the fire and until shutdown cooling was established, the following facts were disclosed:

1. The nuclear cores for both units were adequately cooled during and subsequent to the fire. The minimum water level (Unit 1) was about forty-eight inches above the top of the active fuel. Maintaining reactor water level was never a problem in Unit 2. (Details II, page 7)

TVA Response

This finding is correct.

2. The control room was manned at all times although some smoke and fumes entered the room for a brief period. (Details II, page 5)

TVA Response

This finding is correct.

3. The ability to use the emergency core cooling system (ECCS) as backup to normal cooling was lost for Unit 1. The capability to operate the Unit 1 Standby Liquid Control Systems was lost for at least three hours. (Details II, pages 4 and 8)

TVA Response

The Emergency Core Cooling System (ECCS) is designed to provide adequate core cooling in the unlikely event of a loss of normal core cooling water including a postulated break in the largest reactor cooling water pipe during full power operation. This accident, of course, did not take place. While the availability on unit 1 of emergency core cooling water throughout the event would have greatly facilitated unit 1 core cooling, the reactor was safely shut down using other cooling water sources in conjunction with the operation of the reactor depressurization features of the ECCS. Adequate ECCS for unit 2 was available.

The Standby Liquid Control System (SLCS) is designed to inject boron into the core to shutdown the nuclear reaction if control rods fail to scram (be inserted into the core). The SLCS was not needed at any time during the event as all unit 1 and 2 control rods were observed to have scrammed.

4. Loss of the air supply system for relief valve control resulted from the loss of electrical power to a solenoid valve in the air supply to a diaphragm type isolation valve. This led to the repressurization of the Unit 1 reactor from about 6:40 p.m. to 10:20 p.m. (Details II, page 9)

TVA Response

Overpressure protection was available at all times for both reactors. On unit 1 the capability to blowdown the reactor to low pressure was lost for a period of time for the reason indicated. The valve in question was subsequently bypassed and reactor pressure reduced. The NRC is reviewing a TVA-proposed design change to eliminate this problem in the future.

5. The drywell vent valves for Unit 1 were wired open to prevent drywell pressure buildup. (Details II, page 9)

TVA Response

This was intentionally done to vent the unit 1 primary containment through the charcoal filters of the Standby Gas Treatment System as required by procedure. Because of the fire these valves failed and isolated the unit 1 drywell. However, they were manually opened and wired open to prevent their subsequent inadvertent closure.

6. The spare control rod drive (CRD) pump was inoperable from approximately 1:30 p.m. on. The Unit 2 CRD pump could have been valved to supply Unit 1 if required. (Details II, page 4)

TVA Response

After the fire during a review of engineering drawings, TVA determined that it would have been possible to valve in the unit 2 CRD (control rod drive pump) to the unit 1 reactor. However, the unit 1 pump did not fail and no backup was needed. This item had no impact on the operations on March 22.

V. Effects on Each Unit and Interactions Between Units

During the investigation of the effects on each unit and interactions between units, the following facts were disclosed:

1. The failure of a single 4 KV Shutdown Board results in the loss of power to the ECCS equipment supplied from the board for both Units 1 and 2. This feature is inherent in the design of the electrical systems for Units 1 and 2 and conforms to the FSAR commitments. Unit 3 electrical systems are not shared with Units 1 and 2. (Details III, page 4)

TVA Response

It is important to note that there are two divisions of redundant emergency core cooling equipment for each reactor. Either division can provide adequate core cooling in case of a design basis accident. Failure of a single 4 KV board would affect, at most, equipment in one division on each unit.

This is a well known design feature of the Browns Ferry plant. It was thoroughly reviewed by the Atomic Energy Commission and was approved by the AEC as being completely acceptable.

2. The coordination of the power supplies for the ECCS is such that motors for the systems pumps are not in every case supplied from the same power source as their associated valves. In six cases the pumps and valves are supplied from power sources in the same division, but in two cases the valves are supplied from one division and the pumps from another, in violation of the separations criteria defined in the FSAR. (Details III, page 8)

TVA Response

It is not necessary that pumps and associated valves in the ECCS be supplied from the same power source. The Browns Ferry design is divisionalized such that pumps and associated valves in the ECCS are powered from sources assigned to either division I or II. The loss of pumps and/or associated valves in one division is an acceptable event since either division can adequately cool the reactor in the unlikely event of a postulated design basis accident.

There is an error in the I&E report which apparently has lead to the erroneous conclusion that the criteria were violated.

3. The power cable supplying 480 Volt Shutdown Board 1B from 4 KV Shutdown Board C and the power cable supplying 480 Volt Shutdown Board 2A from 4 KV Shutdown Board B were both routed in the same tray. In that 4 KV Shutdown Board B is in Division I and 4 KV Shutdown Board C is in Division II, this routing constitutes a violation of the TVA separations criteria as defined in the FSAR. (Details III, page 9)

TVA Response

This finding is an oversimplification of a highly technical point. The original design did not assign divisional classification to 4 KV cables between the 4 KV shutdown boards and the 4 KV/480 Volt transformers. The design was analyzed to and met the single failure criterion. Therefore, the routing indicated in the finding was not a violation of the criteria.

4. Power was lost to the common inlet valve for the core standby coolant (raw water) supply for Units 1 and 2. This valve could have been operated manually. (Details III, page 5)

TVA Response

This statement is irrelevant since it ignores the fact that the raw water supply was never required. As an additional backup safety feature, the Browns Ferry design includes provisions for injection of river water into the reactor should it become ultimately necessary in the event all other cooling water supplies fail. As indicated in the statement the inlet valve could have been operated if it had been necessary. In this incident, however, other more preferable water sources were available and were used.

5. Power from the Unit 2 preferred power bus was lost for a period of time because an operator connected it to the Unit 1 preferred bus which sustained faults. A second operator subsequently opened the tie breaker and restored power to the Unit 2 bus. (Details III, page 5)

TVA Response

The unit preferred power buses are not required for safe reactor shutdown. However, they do power annunciation and indicating circuits. The NRC report

indicates that the above item apparently occurred and unit 2 preferred power was affected. However, this was noticed by plant operators and the power supply was restored.

D. Response of TVA Groups and Various Governmental Bodies

During the investigation the following facts were established concerning the response of TVA groups and various state and local governmental bodies:

1. The Athens Fire Department (AFD) responded promptly to the notification of the fire and was prepared to assist within approximately 35 minutes of receiving the alarm. Athens is about ten miles from the site and, upon arrival, the firemen had to be issued personnel radiation monitoring devices prior to entry. (Details IV, page 1)

TVA Response

In accordance with TVA and NRC requirements, personnel must be issued radiation monitoring devices upon entry to the plant.

2. The AFD fire chief initially made the recommendation to use water on the fire at about 2:00 p.m.; however, permission to use water was not given by the Plant Superintendent to the fire fighters until approximately 6:40 p.m. and it was about 7:00 p.m. to 7:20 p.m. when water was actually used. (Details IV, page 1)

TVA Response

The expert fire fighting advice of the Athens Fire Chief was duly considered. However, the plant operators' detailed knowledge of the Browns Ferry design and potential reactor safety implications of the application of water to burning safety-related electrical cables caused them to delay water application to the fire until plant conditions were made more stable. TVA believes in retrospect, after five months of detailed evaluation, that Browns Ferry plant management made the correct decision.

3. The individual who would normally function as TVA Director, Central Emergency Control Center (CECC), was not immediately located. The first alternate was not aware of this and had been at the CECC approximately thirty minutes before realizing that he should have been functioning as Director. (Details IV, page 2)

TVA Response

Although the alternate CECC director did not realize that he was acting as director immediately upon arrival at the CECC, his logs reveal that he, upon notification, contacted the Environs Director and directed him to notify the Alabama Department of Public Health. This delay in recognition did not slow down the Radiological Emergency Plan from being implemented, and the alternate CECC director did carry out the responsibilities of the CECC director for the 30-minute period.

4. Logs kept by individuals at the CECC did not always indicate the times of the events.

TVA Response

TVA believes that all significant or pertinent events were logged at the CECC. Logging of events is a matter of judgment.

5. Information provided by the CECC to others was not always accurate or current. For example, at about 5:00 p.m. an individual at CECC advised NRC that the fire was confined to the spreading room. (Details IV, pages 2 and 3)

TVA Response

The information available to CECC was expeditiously communicated to outside agencies with the full knowledge that, in retrospect, some of the information could have been incomplete. There was no deliberate attempt by TVA to mislead outside agencies.

6. The Director of the CECC was not aware that the fire had been extinguished until approximately 8:15 p.m. when in fact the fire was extinguished prior to 7:45 p.m. (Details IV, page 4)

TVA Response

The fire was extinguished at about 7:45 p.m. *CDT* (Browns Ferry time) and the CECC director in Chattanooga became aware of it at that time. This, of course, is the same as 8:45 p.m. *EDT* (Chattanooga time), which was the time logged in the CECC log provided to NRC.

7. Communications by the CECC with state and local agencies focused on plant operating status rather than offsite radiological releases which is the prime responsibility of these agencies. (Details IV, page 8)

TVA Response

TVA's review of the overall record concludes that CECC communications with outside agencies did focus primarily on the potential for radiological releases. TVA did respond to questions from outside agencies related to plant status with a view toward allowing them to better assess the overall situation.

8. The Alabama equipment for the downwind air sampling was not available for service. Sampling was initiated at Decatur station at approximately 9:00 p.m. on March 22 using a sampler obtained from the Alabama Pollution Control Commission. (Details IV, page 10)

TVA Response

TVA has no comment on responsibilities of the State of Alabama.

9. The State of Alabama emergency plan was out of date, not available to certain responsible officials and some officials were not cognizant of their individual responsibilities. (Details IV, pages 15, 16 and 19)

TVA Response

TVA has no comment on responsibilities of the State of Alabama.

10. Attempts to contact some local officials by state organizations was minimal. (Details IV, page 15 and 16)

TVA Response

TVA has no comment on responsibilities of state organizations.

APPENDIX 15

ATTACHMENTS INCLUDED WITH STATEMENT OF MR. BENJ. RUSCHE, DIRECTOR, OFFICE OF NUCLEAR REACTOR REGULATION, NRC[1]

ATTACHMENT I

NRR EVALUATION OF PLANT SAFETY DURING THE BROWNS FERRY FIRE

Purpose

The purpose of this evaluation is to determine whether alternate methods could have been used to maintain the plant in a safe shutdown condition during the March 22, 1975, fire at the Browns Ferry plant. Such alternate methods are based on the as-built plant design capability. The plant was, in fact, brought to a safe shutdown condition by the operators in a satisfactory manner. At no time was the core uncovered. Our evaluation of the availability of alternate methods of maintaining safe shutdown has been performed to assess the margin of safety that existed throughout the course of the incident.

Approach

1. The evaluations were restricted to Unit 1 since a review of the events made it clear that the effects of the fire on Unit 2 were less, by far, than on Unit 1. At all stages of the incident, Unit 2 had more capability for maintaining a safe shutdown condition. In addition, due to a longer operating history, the decay heat removal requirements were greater for Unit 1.
2. The incident was divided into three specific periods as follows:
   a. Initiation of fire up to reactor scram
   b. Reactor scram, and
   c. Water addition and heat removal .

[1]See p. 77, text.

3. The calculations, which are basic to assessing capabilities of various systems and components in maintaining the core covered and in removing decay heat, are dependent on initial plant conditions and important parameters such as time and water level. Starting points as realistic as possible which were determined by cross checking of facts were used in these calculations.

I. Fire Initiation to Reactor Scram

Shortly after the fire started, spurious events occurred. These events were due to open circuits, short circuits and/or grounds caused by the burning of electric cables. In most cases, the events from the false signals proceeded in the direction of safety. For example, approximately ten minutes following the announcement of the fire, the Emergency Safeguards Systems on Unit 1 received a false signal of a potential loss-of-coolant accident situation. This signal started all emergency core cooling equipment and all four emergency diesel generators. The entire system functioned as designed. No water was admitted to the reactor vessel at this time from low pressure sources since in the absence of an actual coolant loss the reactor vessel remained pressurized. Some water was injected by operation of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. The operator was presented with sufficient information to assure himself that no loss-of-coolant accident condition existed. Since an adequate supply of water to the reactor vessel was available using normal equipment, these emergency systems were shut down. There was some difficulty shutting down these systems initially since the false accident signal was still present. The operator was able to shut down these systems on subsequent attempts.

Other spurious events included various alarms in the control room and conflicting instrumentation readouts. Later in the day, as attempts to bring the Residual Heat Removal (RHR) System into operation were made, a normally-open pump suction valve was found to be closed. It would appear that this valve had been closed by a spurious closure signal.

We have concluded from our investigation that spurious events which occurred in this initial period did not play a particularly important role in shaping the course of subsequent events. However, they were indicative of progressive loss of equipment operability.

Reactor Scram

The Reactor Protection System (RPS) causes automatic reactor shutdown (scram) when preset values of particular variables are exceeded. There are also means for scramming the reactor manually. During the fire, both units were scrammed manually at the judgement of the operator. Automatic scram did not occur prior to this because none of the limits for scram were exceeded. Nevertheless, one may ask whether there was any danger of the scram system being disabled so that the reactor could not be shut down.

Loss of power to the reactor protection system, such as might occur during a fire, is a fail-safe condition that would cause scram. Therefore, the principal question is whether or not the protection system wiring could have been maintained in an energized condition by coming into contact with an extraneous power source when fire damage occurred to cable insulation. All of the reactor protection system cables are run in rigid conduit. No cables containing power

from sources other than the RPS motor generator sets are run in these conduits. Thus, the RPS cables are fully protected by the grounded conduit against the possibility of negating scram by becoming powered by external sources. Heating and loss of insulation on protection system cables within the conduit could lead to short circuits. The results would be blown fuses causing associated relays to de-energize. This is a fail-safe condition leading to reactor shutdown.

We have concluded through detailed examination of the RPS design that the system would have shut the reactor down automatically had it been required because of any transient condition. We have concluded also that fire damage would have inevitably caused automatic scram had the reactor not been shutdown manually.

III. Water Fill and Heat Removal

Starting from an operating condition the normal means of reactor shutdown and cooldown following control rod insertion would be as follows:

a. Continue steam flow to the main condenser through the turbine bypass lines until steam pressure is reduced to a point (less than 100 psig) where condenser vacuum could not be maintained; and,

b. With reactor vessel pressure below 100 psig, initiate the reactor shutdown cooling mode of the RHR system. This system circulates reactor vessel water through a heat exchanger and back to the pressure vessel via an RHR pump.

It should be noted that in the normal method described above, all heat is rejected to outside sources of cooling water, either in the main

condenser or the RHR heat exchanger. No makeup water is required since reactor water is circulated to the heat exchanger and returned to the vessel.

When the Main Steam Isolation Valves (MSIV) closed during the fire, the routing of steam to the main condenser was no longer possible. Due to the loss of operability of certain components by fire damage, the normal means for cooling and making up water for the isolated condition could not be accomplished. The normal means of water fill and heat removal under these conditions would have been as follows:

Decay heat increases reactor vessel pressure to the setpoints of the relief valves which open and provide a flow path to the suppression pool. The decay heat is thus transferred for a short time immediately after shutdown from the fuel to the suppression pool via boiling of the water surrounding the core. This transfer involves a loss of water from the vessel which is automatically made up by operation of either the RCIC or HPCI steam driven turbines which pump water from the condensate storage tanks or from the suppression pool to the vessel. Transfer of heat from the suppression pool to an outside cooling source is accomplished by operating the RHR system in the torus cooling mode during which an RHR pump circulates suppression pool water through a heat exchanger cooled by river water. Shortly after shutdown, decay heat levels are low enough to match the heat removed by steam flow to the RCIC turbine. The relief valves will then close.

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We will now discuss the methods utilized by the operator and alternate methods that were available. For reference purposes, Table 3 provides a listing of all plant pumps normally capable of providing makeup water to the pressure vessel, including both high pressure and low pressure capability. The availability of these pumps during the incident is covered under the remarks section of the Table.

The discussion can be separated in terms of water makeup and heat removal. Heat removal from the fuel was accomplished automatically through relief valve operation and steam release to the suppression pool. The large volume of the suppression pool gives it a very large heat absorption capacity, and 10 to 12 hours of decay heat can be handled prior to the need for operator action. Heat removal to an ultimate sink is not as critical a consideration as water makeup and will, therefore, be discussed later.

At high vessel pressure, the automatic makeup is normally provided by the feedwater system backed up with either the steam driven HPCI or RCIC systems. Neither the HPCI or RCIC were automatically available following spurious operation at the start of the fire.

With the only source of high-pressure makeup water in operation being a single control rod drive (CRD) pump, the plant operators decided to reduce the pressure in the vessel to a point where they would be able to provide makeup water . low pressure equipment known to be operable. However, other equipment could have been used to maintain the core covered with water.

Besides the CRD pump on Unit 1, other installed sources of high pressure makeup were the CRD pump on Unit 2, a shared spare CRD pump and standby liquid control (SLC) pumps. The CRD pumps, while performing their normal functions associated with the control rod drive system, also provide water to the vessel while at any pressure. One CRD pump is normally in operation per unit and the pump for Unit 1 operated continuously throughout the course of the incident. In addition the SLC pumps are each capable of providing approximately 56 gpm at pressures up to reactor coolant system design pressure. The SLC pumps were not required as a backup reactivity shutdown system since the control rods functioned normally. However, there was a time period of up to three hours following the initiation of the fire during which they were not available due to loss of power. Means of utilizing installed sources of make up water and an analysis of their effectiveness follows. The effectiveness of a given component can be illustrated by utilizing Figures 1 & 2. These figures show the water makeup rate required to keep the core covered as a function of time after shutdown. Their bases are detailed with the figures.

Although our calculations are based on the core covered with water it should be pointed out that this is not necessary to prevent serious core damage or melting. If the core uncovers, steam generated in the portion of the core that is covered will still provide some cooling to the upper and uncovered portion of the core.

Fuel rod cladding temperatures could be allowed to reach at least 1800°F without serious consequences. Therefore, the steam leaving the core could

be at nearly 1800°F rather than at saturation temperature. The increase in temperature would increase the enthalpy or heat content for the same mass flow, and thus a lower flow would be adequate. However, since the core remained covered no heatup beyond saturation temperature was experienced during or following the fire.

The CRD pump in operation was part of a system for Units 1 and 2 which consisted of three CRD pumps. One pump normally operates for each unit and the third pump can be used on either unit. Subsequent examination of the actual piping configuration confirmed that it is also possible to align the Unit 2 pump to provide water to Unit 1. Means also exist to increase the output of a CRD pump by valving in a pump test bypass line which provides an additional flow path. It is estimated that by opening this single valve it would have been possible to have provided sufficient water, approximately 225 gpm, to maintain the core covered throughout the course of the incident. No other systems would have been required to provide water to maintain an adequate inventory of water in the reactor vessel and depressurization would not have been necessary. This flow (225 gpm) could have been increased to in excess of 300 gpm with an additional CRD pump. Figure 1 shows that a flow rate of approximately 218 gpm would have been adequate to maintain core coverage throughout the incident.

An additional source of high pressure water mentioned previously as being unavailable due to fire damage was the RCIC system. It would have been capable of providing sufficient flow (600 gpm) for makeup water requirements throughout the entire course of the incident if the operators'

decision had been to make it available. [illegible]ears that this system could have been made available within an hour [illegible]king this decision. The source of steam for the RCIC system would have been the auxiliary boiler which was used for testing the RCIC prior to plant operation. Two procedures are necessary to provide the steam path. First, the auxiliary boiler must be put into operation. Full steam pressure from this source can be obtained in less than one hour. (The operators actually put the auxiliary boiler into operation later in the incident). The second procedure is the installation of a spool piece to make up the flow path from the auxiliary boiler to the RCIC turbine. This could have been accomplished in less than one hour. The operation of the RCIC would then have been possible from the backup control room where there is control and indication of system operation both in terms of water flow to the vessel and turbine speed.

Beyond the RCIC system, there remains the standby liquid control (SLC) pumps mentioned previously. If the decision had been made to make the pump capacity available as high pressure makeup, the operators have indicated that these pumps could have also been made available within an hour. The capacity of these pumps alone would not have been sufficient

to provide for complete water requirements. The capacity of these pumps along with the flow rate being obtained from the operating CRD pump (total of approximately 212 gpm) would have provided a sufficient quantity of water to keep the core adequately cooled. Obviously, the use of one or both of the SLC pumps in conjunction with other makeup sources already noted would have provided more than an adequate supply of high pressure water to assure that the core would have remained covered.

We have discussed above various alternate means of obtaining high pressure sources of makeup water which were not utilized. Instead, as we noted previously, a decision was made by the operators to depressurize the reactor. This was accomplished by using the four relief valves available for remote manual operation from the control room. Only one would have been required to perform the function. The remaining seven relief valves with remote manual capability were not operable because of fire damage to the power supplies to the solenoids controlling air flow to the valves. However, it has been established that should the operators have needed any of the other valves it would have been practical to supply solenoid opening power at the appropriate electrical connecting pins at the containment penetration.

Once the reactor vessel pressure was reduced to 350 psig one condensate booster pump and one condensate pump provided an adequate source of makeup water. In addition the operators had available two additional condensate booster pumps and two additional condensate pumps. Any one of the condensate pumps was capable of supplying more than enough water to maintain the core covered at reactor pressure below 160 psig. An additional source of low pressure water for injection into the reactor vessel was river water. If needed, either of two available pumps could have provided sufficient river water to keep the core covered as long as the reactor pressure was down to 185 psig or less.

During the stable low pressure operation in which the condensate booster pump and condensate pump were providing makeup water, operability of the relief valves being used for maintaining low pressure in the vessel was lost as a result of the loss of control air. At this time the relief valves closed and the continued production of decay-heat resulted in a pressure increase in the vessel. The pressure increased beyond the capability of the available low pressure injection systems, and the capability of providing water by these systems was lost. The operators established the cause of the loss of control air and the decision was made to bypass the closed solenoid valve in the common relief valve air supply line. The air supply in the accumulators designed to provide for relief valve operation under isolation conditions had been depleted.

The bypass action was accomplished in approximately 3-1/2 hours following which time the reactor vessel pressure was once again reduced within the capability of the low pressure condensate booster pump and the condensate pump.

There were other courses of action which might have been taken by the operator at the time that remote-manual operability of the relief valves was lost. No immediate problem existed since the pressure would have increased up to the setpoints of the relief valves in their over-pressure protection mode with subsequent steam relief to the suppression pool. The CRD pump was providing a source of makeup water. With the much reduced decay heat, considerable time was available for other operator action. The alternatives discussed under high pressure makeup were still available if control air to the relief valves could not be reestablished.

Calculations, however, indicate that no operator action was necessary to maintain the plant in a safe condition. This is due to the availability of a depressurization and heat removal path via the main steam line drain valves to the condenser. Both of these valves were inoperable

by electrical means as a result of fire damage. The operators however, had decided previously to return draining capability to the main steam line and this had been achieved approximately one hour after control air to the relief valves was lost. It is calculated that the quantity of steam being removed from the pressure vessel through the main steam drainline was great enough that the reactor pressure would have leveled off at a point prior to reaching the relief valve set-point. An equilibrium condition would then have been maintained with the reduced reactor pressure reducing the head on the operating CRD pump such that the pump would provide sufficient makeup flow to maintain the core covered throughout the remainder of the incident.

The calculations providing the basis for these statements are as follows:

1. Maximum steam flow rate through the 4 inch drain line during pressurization of the reactor (between 7:30 and 9:30 PM) based upon an average pressure of 530 psig is 122,000 lb/hr.
   a. Adiabatic compressible flow with friction Fanno line with choking,
   b. from observation of the piping: $4(\frac{fl}{D}) = 20$,
   c. inlet mach number = .185, and
   d. inlet velocity = 328 ft/s.

However, the loss of 122,000 lbs of water would cause the reactor vessel liquid level to decrease 114 inches in one hour,

$$\frac{122{,}000 \text{ lb/hr}}{50 \text{ lb/ft}^3 \times 20.9 \text{ ft/inch of reactor level}} = 114 \text{ inches/hr}$$

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Since the reactor liquid level remained essentially constant through the prepressurization period it is obvious that the actual steam flow was considerably less than the calculated as a maximum. However, the flow would necessarily have been equal to or greater than the makeup flow provided during that time period by the CRD pump. This is based on the level not showing a net increase during the time period even though the CRD pump was continually in operation. We can therefore conservatively state that the drain capacity at the average 530 psig pressure was 152 gpm corresponding to 76,000 lb/hr. The 152 gpm is estimated from the CRD pump head and system resistance curves.

2. From the reactor pressure recorder stripchart it was determined that had the relief valve not opened, the pressure rise. extrapolated to about 2:00 AM would have reached 700 psig; however, at 700 psig the steam discharge rate would have increased, i.e.,

$$\frac{\text{Steam discharge rate @ 700 psig}}{\text{Steam discharge rate @ 530 psig}} = \frac{@\ 700}{530}\sqrt{\frac{T_{Abs\ 700}}{T_{Abs\ 530}}}$$

$$= \frac{.86}{.64} \times \sqrt{\frac{965}{934}} = 1.34$$

at 700 psig, steam discharge rate = 102,000 lb/hr X 1200 BTU/lb. This corresponds to a heat load of 122 X 10^6 BTU/hr, i.e., 102,000 lb/hr X 1200 BTU/lb. However, at 2:00 AM (about 47,000 seconds after scram) the reactor deay heat is 0.715% of full power = 23.6 MW which corresponds to 80 X 10^6 BTU/hr. Since for the assumed pressure of 700 psig, more energy leaves the reactor than that which is produced, the reactor pressure could never get as high as 700 psig.

Furthermore, the decay heat rate of 23.6 MW corresponds to a makeup rate of 142 gpm, which is achievable with one CRD pump operating against a reactor pressure of 660 psig (at 700 psig 1 CRD pump will deliver ~ 139 gpm).

The operators however did regain use of control air and proceeded to depressurize the reactor vessel which permitted use of low pressure pumps. This stable condition continued until normal means became available (which occurred only following clearing of the smoke sufficiently for the operators to establish the proper valve lineup) such that torus cooling could be attained. Thereafter a normal reactor shutdown cooling mode was used.

The means of heat removal available through the course of the incident was not critical due to the availability of relief valves to transfer the decay heat to the suppression pool which is of sufficient volume to preclude the need for any operator action for a time period of 10 to 12 hours following the start of the incident. The suppression pool could then be permitted to boil, if necessary, with the steam vented through the standby gas treatment system to the atmosphere.

There were also other means of heat removal from the reactor. Once the main steam line drain valves were open, a flow path existed to the condenser whereby the decay heat was being transferred to the river water. This greatly reduced the heat load on the suppression pool within the torus and eventually would have permitted removal of all of the generated decay heat. Means of removing the decay heat from the torus prior to availability of normal torus cooling existed. There are several

flow paths by which water could be drained from the torus thereby removing the heated water and by which cold makeup water would be introduced to the torus. The reactor heat removal drain pumps were operable and provided flow paths by which water could be transferred from the torus to the main condenser and also from the condensate storage tanks to the torus. Flow rates far in excess of those required were available. The staff concludes that alternate means of heat removal were available throughout the course of the incident. Normal suppression pool cooling was established 12 hours after scram and normal shutdown cooling of the reactor was established 15 hours after scram.

The core remained covered throughout the incident. The operators always had a course of action which they were pursuing and in addition there was always at least one or more redundant means available to cool the core.

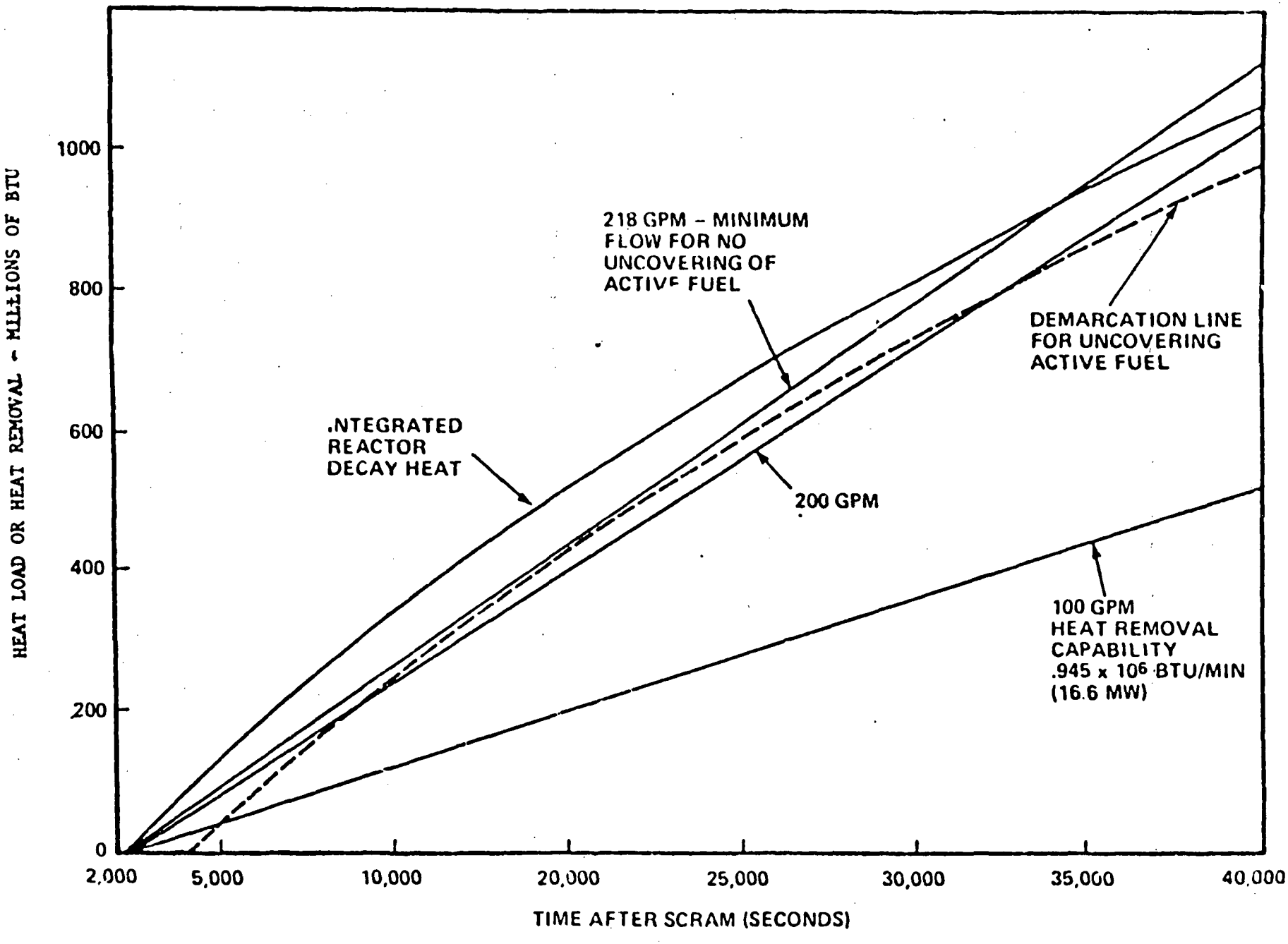
HEAT LOAD OR HEAT REMOVAL - MILLIONS OF BTU
1000
800
600
400
200
0
2,000
5,000
10,000
20,000
25,000
30,000
35,000
40,000
TIME AFTER SCRAM (SECONDS)
218 GPM - MINIMUM FLOW FOR NO UNCOVERING OF ACTIVE FUEL
DEMARCATION LINE FOR UNCOVERING ACTIVE FUEL
INTEGRATED REACTOR DECAY HEAT
200 GPM
100 GPM HEAT REMOVAL CAPABILITY .945 x 10^6 BTU/MIN (16.6 MW)

Figure 1.

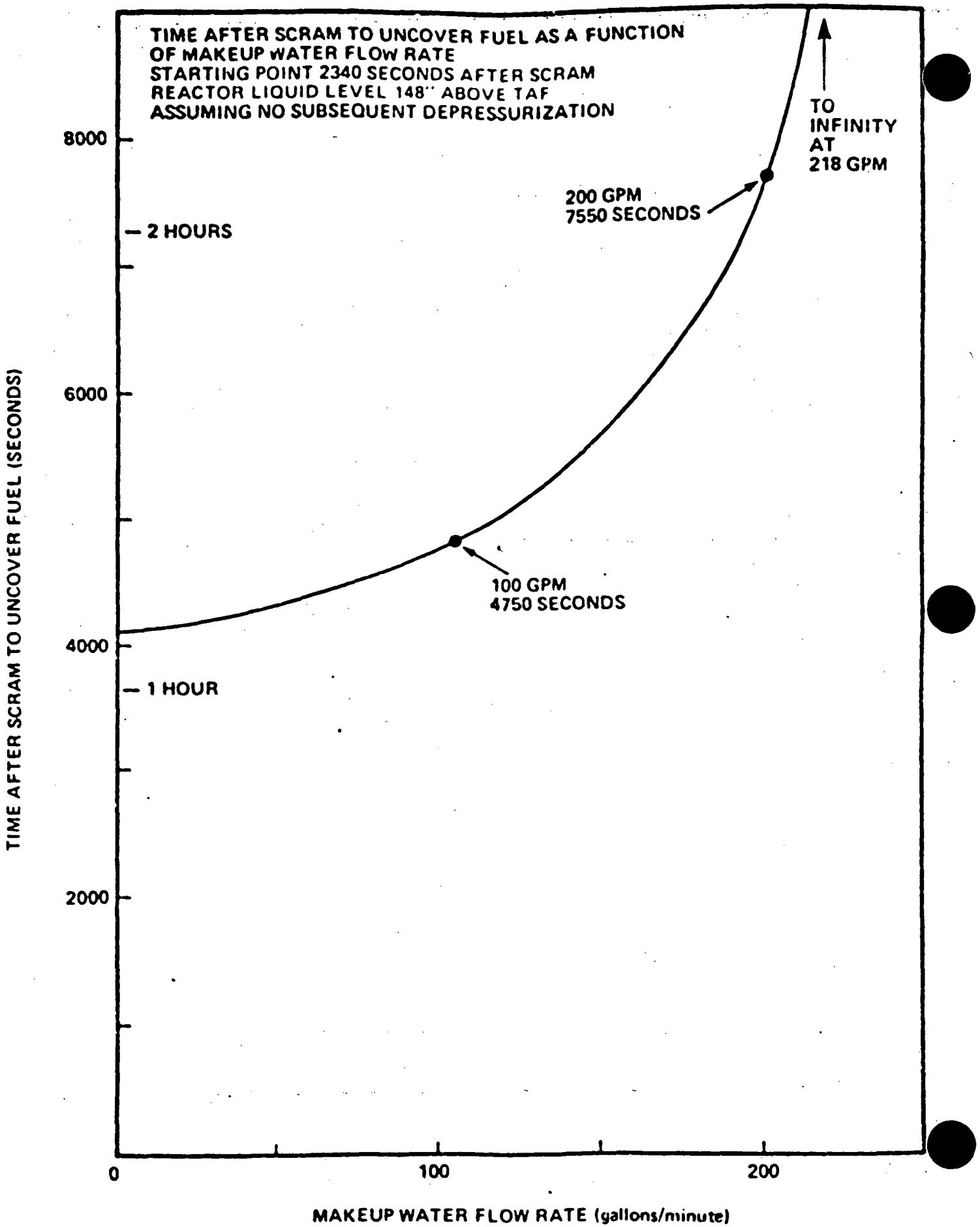
TIME AFTER SCRAM TO UNCOVER FUEL AS A FUNCTION
OF MAKEUP WATER FLOW RATE
STARTING POINT 2340 SECONDS AFTER SCRAM
REACTOR LIQUID LEVEL 148" ABOVE TAF
ASSUMING NO SUBSEQUENT DEPRESSURIZATION
TO INFINITY AT 218 GPM
200 GPM 7550 SECONDS
100 GPM 4750 SECONDS
2 HOURS
1 HOUR
8000
6000
4000
2000
0
100
200
TIME AFTER SCRAM TO UNCOVER FUEL (SECONDS)
MAKEUP WATER FLOW RATE (gallons/minute)

Figure 2.

BASES FOR DEVELOPMENT OF FIGURE 1

Figure I presents the results of calculations which were made to determine the amounts of coolant which had depressurization not been accomplished about 40 minutes after scram, would have been necessary to keep the fuel covered.

The line labeled reactor decay heat represents the amount of decay heat released by the core starting approximately 2340 seconds after reactor scram. This curve was obtained by numerical integration of the ANS decay heat curves (Shure + actinides + uncertainty), corrected for finite operation (equivalent of 280 full power days). The values used for the numerical integration appear in Tables 1 and 2.

The lines in figure 1 which are labeled 100, 200, 218 gpm show the amount of heat which could be removed by introduction of coolant (at 100, 200, and 218 gpm), starting at 2340 seconds. For these calculations it is assumed that water is introduced at 90°F, and is exhausted as saturated vapor at 1080 psig. Thus a continuous flow of 100 gpm can remove 16.6 mw of heat (945,000 BTU/min)--note that the 100 gpm line is constructed by connecting a straight line through the points: x = 2340 seconds, y = 0 BTU, and x = 8340 seconds, y = 94.5×10^6 BTU. The lines of 200 and 218 gpm are constructed similarly.

At 2340 seconds, the water level observed in the reactor was 148 inches above the top of the active full (TAF); this corresponds to 3095 ft^3 of water. Since the initial condition of the vessel was

one of saturation at 1080 psig, the density of the water in the reactor vessel was - "5.5 lb/ft and the latent heat of vaporization for the water at that pressure was 631 BTU/lb. The amount of decay heat necessary necessary to boil off the initial volume of water was:

$$148 \text{ in. above TAF} \times \frac{20.9 \text{ ft}^3}{\text{in. of reactor water level above TAF}} \times 45.5 \text{ lb/ft}^3 \times 631 \text{ BTU/lb} = 89 \times 10^6 \text{ BTU}$$

Thus, if there were no makeup water provided, the time at which the core would become uncovered is = 4050 seconds after scram (point on the decay heat curve where 89×10^6 BTU are liberated from the core).

If, however, makeup water at the rate of 100 gpm were initiated at time 2340 seconds after scram, at time 4050 seconds after scram the makeup water would have been able to account for 27.5×10^6 BTU of decay heat and the core would not have been uncovered. The difference between the integrated decay heat, and the makeup water cooling capacity represents the net amount of heat used to boil off the initial reactor vessel inventory.

Proceeding to time = 4750 seconds after scram, the integrated decay heat is = 127.5×10^6 BTU, and the heat removal capability of the makeup water introduced at 100 gpm is = 38.5×10^6 BTU. The difference between these values is 89×10^6 BTU, enough to uncover the active fuel. Thus, the dashed line - labeled "Demarcation line for uncovering the active fuel" was plotted 89×10^6 BTU below the integrated reactor decay heat curve. Therefore, introducing 100 gpm of coolant will assure that the active fuel is covered until 4750 seconds after scram.

Similarly, introducing coolant at a rate of 200 gpm will assure that the active fuel is covered until ≈7520 seconds after scram (note at 7520 seconds after scram the 200 gpm line is 89 x 10^6 BTU below the integrated decay heat curve (7520 seconds, 162.5 x 10^6 BTU, is the point of intersection of the 200 gpm line and the dashed "Demarcation" line).

Similarly, examination of figure 1 reveals that the 218 gpm line will never cross the demarcation line; consequently, introduction of coolant at 218 gpm will prevent the core from ever being uncovered.

Summarizing these findings -- starting at a reactor pressure of 1080 psig with a liquid level of 148 in. above TAF, at 2340 seconds after scram, introduction of cooling water at 90°F will result in the following:

Coolant Rate, Gallons/Minute	Time After Scram To Uncover Fuel, Seconds
0	4050
100	4750
200	7550
218	∞

Figure 1 is a convenient tool for determining the makeup water's fuel covering capability for the aforementioned event.

Figure 2 is a cross plot of Figure 1 to directly show the time from scram to core uncovering as a function of makeup coolant flow rate.

DECAY HEAT CALCULATION

From TVA's plan for Evaluation Repair and Return to Service Part VI Section E page 59 (5/28/75) the reactor had 280 full power days prior to the fire. This equates to,

$$280 \text{ days} \times 86{,}500 \frac{\text{sec}}{\text{day}} = 24 \times 10^6 \text{ seconds}$$

which according to the Shure decay heat curve represents a 0.085% full power correction term (to account for the fact that the reactor had not been in operation for infinite time).

The heat removal capability of cooling water:

assume, $T_{in} = 90°F$, $h_f = 58$ BTU/lb

$P_{out} = 1080$ psig, hg = 1189 BTU/lb

$$1 \frac{\text{gallon}}{\text{minute}} \times .1337 \frac{\text{ft}^3}{\text{gallon}} \times \frac{\text{lb}}{.016 \text{ ft}^3} \times (1189 - 58) \frac{\text{BTU}}{\text{lb}}, \text{ or}$$

$$1 \text{ gpm} = 9450 \frac{\text{BTU}}{\text{min}}, \text{ which is equivalent to, } 0.166 \text{ MW.}$$

TABLE 1

REACTOR DECAY HEAT

Time After Scram (seconds)	Decay Heat % of Full Power	Interval Avg. % of Full Power	Total Interval Heat ($x10^6$ BTU)
40	5.015		
100	4.215	4.615	8.66
160	3.78	3.997	7.52
220	3.53	3.655	6.86
300	3.265	3.398	8.51
400	3.06	3.16	9.9
600	2.765	2.913	18.2
660	2.665	2.715	5.1
1000	2.5	2.583	27.4
1300	2.1	2.3	21.6
1900	1.99	2.045	38.4
2500	1.76	1.875	35.3
4300	1.59	1.67	94.5
6100	1.44	1.515	85.5
7900	1.3	1.37	77.3
9100	1.2	1.25	47.0
15,000	1.02	1.11	209.0
21,100	.915	.965	181.0
27,100	.815	.865	163.0
39,100	.715	.765	288.0
51,100	.7	.707	265.0

1% fp - sec = $3.13x10^4$ BTU

Table 2

Decay Heat Subsequent to Time of

First Reactor Depressurization 2340

Seconds After Scram (Water Level 148" above TAF)

Time Sec	Decay Heat per Interval (BTU x 10^6)	Decay Heat Integral (BTU x 10^6)
2340		
2500	8.95	9
4300	94.5	103.5
6100	85.5	189
7900	77.3	266
9100	47	313
15,100	209	522
21,100	181	703
27,100	163	866
39,100	288	1154
51,000	265	1419

TABLE 3

PLANT SOURCES OF MAKEUP WATER

	Number Provided	Flow rate for each Pump	Comments
Control rod drive pumps	1 per unit plus 1 shared spare All 3 could have been aligned for 1 Unit	Normal operation: 100 gpm at reactor pressure of 1080 psig 182 gpm at reactor pressure of 0 psig With pump test: bypass line open 225 gpm at reactor pressure of 1080 psig	Available throughout event (used in normal operating mode)
Standby liquid control pumps	Two	56 gpm each at reactor pressure of 1080 psig (positive displacement pumps)	Available for majority of post scram conditions (powerless for several hours due to fire damage. Could have been energized in less than 1 hr. if needed.
Reactor Core Isolation Cooling Pump (Steam turbine driven)	One	600 gpm at reactor pressure of 1080 psig	Unavailable due to damage to connecting wiring-however could have been made available in less than 1 hour if needed

Table 3 (continued)

Plant Sources of Makeup Water

	Number Provided	Flow Rate for Each Pump	Comments
High pressure Coolant Injection Pump (Steam turbine driven)	One	5000 gpm at reactor pressure of 1080 psig	Unavailable due to fire damage
Feedwater Pumps (Steam turbine driven)	Three	11200 gpm	Unavailable due to closure of main steam isolation valves, inadequate steam supply
Condensate and Condensate Booster Pumps	Three	10800 gpm (low pressure < 250 psig)	Available throughout event - could only be used in low pressure mode
Condensate Pumps	Three	10800 gpm (low pressure)	Available throughout event - could only be used in low pressure mode
Low Pressure Coolant Injection Pumps	Four	10000 gpm (low pressure)	Two unavailable due to fire damage to cables
Core Spray Pumps	Four	3125 gpm (low pressure)	Two unavailable due to fire damage to cables
Residual Heat Removal Service Water Pumps	Two	4500 gpm (low pressure)	Both available

ATTACHMENT II

LETTER FROM THE NRC TO THE TVA DATED MAY 9, 1975, WITH ENCLOSURES

AMENDMENT NO. 9 TO LICENSE NO. DPR-33

AMENDMENT NO. 6 TO LICENSE NO. DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 9, 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 9 to Facility License No. DPR-33 and Amendment No. 6 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Units 1 and 2. These amendments include Change No. 10 to the respective Technical Specifications and are in response to your request dated May 8, 1975.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the two units will remain in a safe and stable posture during the shutdown time interval between now and our subsequent authorization for the start of repair and restoration of the facility.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 9 to License No. DPR-33
2. Amendment No. 6 to License No. DPR-52
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 9
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 8, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-33 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 10 to Technical Specifications

Date of Issuance: May 9, 1975

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated May 8, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-52 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 10."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 10 to Technical Specifications

Date of Issuance: May 9, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 9 TO LICENSE NO. DPR-33

AND LICENSE AMENDMENT NO. 6 TO LICENSE NO. DPR-52

(CHANGE NO. 10 TO THE TECHNICAL SPECIFICATIONS)

DOCKET NOS. 50-259 AND 50-260

Revise Appendix A as indicated.

FACILITY OPERATING LICENSES DPR-33 AND DPR-52

TEMPORARY TECHNICAL SPECIFICATIONS AND BASES

FOR

BROWNS FERRY NUCLEAR PLANT

UNIT 1 AND UNIT 2

INTRODUCTION

As a result of equipment damage resulting from the fire which occurred at the facility on March 22, 1975, various technical specifications set forth in the technical specifications governing operation,[1] are no longer appropriate. For example, certain items are not "operable" as defined in the technical specifications governing facility operation, but will nontheless perform the functions necessary to assure plant safety. These Temporary Technical Specifications and their respective bases will provide assurance that the plant equipment and systems in their present status and configurations are adequate to provide for plant safety for the work activities being carried out at the facility during the period in which these specifications are effective.

Certain work is planned for the period during which these specifications are to be effective. To assure that certain components, which will be required to function during restoration and repair of the facility (which is not covered by their specifications) are protected during this period, some components will be rerouted and reconnected to obtain a configuration which is protected during repair. Also, certain instruments or components will be changed from automatic operation to manual operation or will be locked in a safe configuration. These modifications will be performed during the period in which these temporary specifications are effective.

However, during the period in which these specifications are effective, it is intended that no intentional action will be taken which would entirely remove a safety system from service except for essential maintenance. To assure this, no intentional action will be taken to cause components required to achieve the objectives of these technical specifications to lose their functional capability except for maintenance

[1] The term "technical specifications governing facility operation" means Facility Operations Licenses DPR-33 and DPR-52, Technical Specification and Bases for Browns Ferry Nuclear Plant Unit 1 and Unit 2, Limestone County, Alabama, Tennessee Valley Authority, Docket Nos. 50-259, 20-260, effective prior to May 5, 1975.

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required to prevent a substantial increase in the risk of failures to the component or any other system or component presently functional in the facility. However, automatic features in components may be altered when such actions enable such components to be operated manually or locked in a safe configuration to achieve the status specified for such component in the "TVA Safety Analysis of the BFNP Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduit," dated May 1, 1975, and included as Part 6, Section E, of the Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire).

These temporary technical specifications will be effective immediately upon issuance and will remain effective until superseded. During the period in which these technical specifications are effective, each reactor will be maintained in cold shutdown. In this condition, the mode switch shall be locked in shutdown, the reactor coolant temperature will be equal to or less than 212° F, and the reactor will be maintained in the vented position.

These temporary specifications do not authorize removal of the fire-affected features or restoration of the facility. Before such activities are authorized, these temporary specifications will be superseded in the whole or in part.

These temporary specifications may otherwise be supplemented, amended, or modified from time to time.

The temporary specifications set forth in this addendum, are in the form of specified sections, subsections, or page changes as identified on the particular change. These changes are temporary only.

1.0 DEFINITIONS

This technical specification (pages 2-7) remains unchanged.

SAFETY LIMIT

1.1/2.1 FUEL CLADDING INTEGRITY

Delete present section (pages 8-25) in its entirety and replace with the following:

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Manual operation under procedural controls shall be established to assure the reactor water level remains ≥378 inches above vessel zero.

1.1/2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The objective of this specification is to assure that irradiated fuel in the reactor vessel remains covered with water at all times. When plant equipment as specified in facility technical specification is available, this objective is accomplished automatically. This is no longer possible. Accordingly, procedural control of the water level is required as set forth in the specification 3.2.A.1.

With the reactor in cold shutdown, rapid makeup of coolant inventory is not required. The provisions for manual operation under procedural controls provided by the "TVA Safety Analysis of the BFNP Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduit," dated May 1, 1975, and included as Part 6, Section E, of the Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire), will assure timely isolation of leaks by requiring redundant level alarms and indicators; double isolation features, one of which shall be operative from the control room, will prevent draining of the vessel as a result of any single active component failure. Also, in the unlikely event that coolant inventory is required, a core spray pump, capable of delivering rated flow, will be available to provide the required vessel makeup capability.

The limiting safety systems scram setpoints are not capable of performing their intended function since the automatic actions they would initiate are unavailable. However, the facility has been placed in a condition that prevents the need for these actions.

1.2/2.2 REACTOR COOLANT SYSTEM INTEGRITY

Delete present specification in its entirety (pages 27-30) and replace with the following:

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SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the protective action to prevent the pressure safety limit from being exceeded.

Specification

A. The reactor vessel head vent valves shall be kept in the open position.

B. Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards.

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1.0/2.2 BASES: REACTOR COOLANT SYSTEM INTEGRITY

With the reactor in the cold shutdown condition and with redundant RHR shutdown cooling capability, pressurization of the reactor vessel is prevented. In the unlikely event that shutdown cooling capability from the RHR system is lost, manual operation of any one of the four available relief valves prevents pressurization of the reactor pressure vessel above 50 psig preserving the safety limit. This keeps the pressure low enough to permit addition of water by a core spray pump.

Checks of the reactor head vent and relief valve position indicating lights each shift confirm control power available for actuation.

3.1/4.1 REACTOR PROTECTION SYSTEM

Delete present specification (pages 31-49) in its entirety and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system.

Specification

A. The setpoints and minimum number of channels required to be functional are listed in Table 3.1.A/4.1.A.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrument channels shall be functionally tested and calibrated as indicated in Table 3.1.A/4.1.A.

TABLE 3.1.A/4.1.A

REACTOR PROTECTION SYSTEM INSTRUMENTATION

Minimum Number of Functional Channels per Trip System	Trip Function	Trip Level Setting	Functional Test Frequency	Alarm Level Setting	Calibration Frequency
1	Manual scram	N/A	once/month	N/A	N/A
2 (1)	SRM high-high count	$\leq 5 \times 10^5$ cps	once/week	$\leq 5 \times 10^5$ cps (upscale high high) $\leq 10^5$ cps (upscale high) ≥ 3 cps (downscale)	once/3 months

Notes for Table 3.1.A/4.1.A

(1) The SRM's are presently connected in the non-coincident mode in the RPS, where any 1 SRM upscale high high will initiate a scram. Of the 4 SRM channels, only 2 are required to be functional.

1.1/4.1 BASES: REACTOR PROTECTION SYSTEM

The reactor protection system will be available for rod insertion in the unlikely event that a rod becomes withdrawn. This is highly improbable because the reactor mode switch is locked in the shutdown position and the control rod directional solenoids are electrically disarmed. The reactor protection system is functional as described in Section 7.2 of the FSAR with the manual scram and the SRM's required functional. The SRM's have been placed in the non-coincidence scram mode to provide automatic scram from high-high count rate.

3.2/4.2 PROTECTIVE INSTRUMENTATION

Delete present specifications (pages 50-107) in its entirety and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the plant instrumentation which initiates and controls a protective function.

Objective

To assure the operability of protective instrumentation.

Specification

A. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information required in the period during which these specifications are effective are given in Table 3.2.A.

1. Reactor water level shall be maintained at ≥ 555" above vessel zero.

SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specification

A. Surveillance Instrumentation

The surveillance requirements associated with the equipment listed in Table 3.2.A shall be that indicated in Table 4.2.A.

1. The reactor water level shall be verified once each shift.

TABLE 3.2.A
SURVEILLANCE INSTRUMENTATION
(POST ACCIDENT INSTRUMENTATION)

Minimum No. of Operable Instrument Channels	Instrument No.	Instrument	Type Indication and Range	Alarm Setting	Notes
2	LI-3-206 or LR-3-53 or LI-3-55	Reactor Water Level	Indicator 0" to 60" Recorder 0" to 60" Indicator 0" to 60" Indicator 0" to 400"	Low $\geq$ 27", High $\leq$ 39"	(1)
2	PI-3-54 PR-3-53	Reactor Pressure	Indicator 0-1200 psig Recorder 0-1200 psig	High $\leq$ 1040 psig	(1)
2	PR-64-50 and PI-64-67	Drywell Pressure	Recorder 0-80 psig Indicator 0-80 psig		(1)
2	TI-64-52A and TR-64-52	Drywell Temperature	Indicator 0-400°F. Recorder 0-400°F.	High $\leq$ 145°F	(1)
2	TI-64-55A and TIS-64-55	Suppression Chamber Water Temperature	Indicators 0-400°F.	High $\leq$ 90°F.	(1)
1	LI-64-54A	Suppression Chamber Water Level	Indicator -25" to +25"		(1)
1	NA	Control Rod Position	Continuity		(2)
2	SRM A, B, C, D	Neutron Monitoring	Indicator and Recorder 0.1 to 10^{+6} cps -100 to +10 sec (period)	Downscale $\geq$ 3 cps Retract permit $\geq$ 100 cps Upscale High $\leq 10^5$ cps Upscale High-High $\leq 5 \times 10^5$ cps Period $\geq$ 30 sec.	(1) (3)

NOTES FOR TABLE 3.2.A

(1) If one of the instrument channels monitoring a parameter should become incapable of performing its intended function, all operations which could affect the associated system will be suspended until the item is returned to service.

(2) The control rod position indicator full-in switches will be operable for every control rod and will provide indication in the control room, or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel.

(3) The following Source Range Monitoring Channels will be operable and will provide count rate indication and alarms in the control room:

SRM Channel A or C

SRM Channel B or D

The alarms in the control room will be as follows:

SRM TRIPS

Trip Function	Trip Action
SRM upscale or inoperative	Annunciator, amber light
SRM Downscale	Annunciator, white light

In addition, the SRM's have been placed in the non-coincidence scram mode to provide for protection against high neutron flux.

TABLE 4.2.A

MINIMUM TEST AND CALIBRATION FREQUENCIES FOR SURVEILLANCE INSTRUMENTATION
(POST ACCIDENT INSTRUMENTATION)

	Instrument Channel	Calibration Frequency[3]	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Level	Once/6 months	Once/Week (4)
7)	Control Rod Position	NA	Each Shift[1]
8)	Neutron Monitoring (SRM)	Once/3 months[2]	Each Shift
9)	Drywell Pressure (PS-64-67)	Once/6 months	NA

NOTES FOR TABLE 4.2.A

(1) Control rod position indication to verify the full-in position will be maintained on unit 2. Unit 1 control rods which do not have position indication in the control room will be monitored at the local panel by use of the full-in limit switches at intervals not to exceed once/3 days.

(2) The SRM will be functionally tested on frequency of once per week.

(3) Instrument calibration will be verified upon restoration, then at the frequency listed.

(4) Instrument check will be to observe instrument response as transmitter is removed and returned to service. A redundant instrument is not available for comparison.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION

For each parameter monitored, as listed in Table 3.2.A, there are equivalent or redundant channels of instrumentation as noted. By comparing readings between two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will necessitate an early recalibration, thereby maintaining the quality of the instrument readings.

The redundant monitoring of reactor vessel level will be sufficient to provide the operator with information with which to determine the adequacy of the coolant inventory. In addition, overfilling of the reactor vessel will be detected by verifying that reactor vessel water level is not rising to an abnormally high level. The approach of abnormal conditions will be brought to the operator's attention by audible alarms.

Instrumentation has been selected to provide control room indication of adequate information regarding status of the drywell pressure and temperature and torus water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

A minimum of two channels of the SRM's, each on separate power supplies, will be operable. Each channel will provide indication of count rate and provide alarms as listed in Table 7.5-1 of the FSAR.

An automatic scram from high-high count rate of any SRM's channel as well as a manual scram will be available. One control rod drive pump each for units 1 and 2 will be kept in operation and all the accumulators will be kept charged with nitrogen and water. The scram pilot valves and pneumatically controlled valves to scram each control unit will be operable.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION (CONTINUED)

Control rod position indication to verify the full-in position will be maintained on unit 2. The continuity of all the full-in limit switches on unit 1 will be monitored periodically. Under the conditions to be maintained in Temporary Technical Specification 3.3, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions.

The reactor water level will be nominally maintained at a level greater than 27" but less than 39" as indicated on LR-3-53. Above or below this level will give an alarm of "Reactor Water Level Abnormal." 27" indication on LR-3-53 corresponds to 555" above vessel zero.

3.3/4.3 REACTIVITY CONTROL

Delete this specification in its entirety (pages 108-121) and replace with the following

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity.

Specification

A. All control rods shall be inserted in the full-in position.

B. The directional control valves shall be disarmed electrically for all control rods.

C. The manual valves in the drive water supply shall be in the shut position to prohibit rod movement.

D. The control rod accumulators shall be charged.

E. Two SRM channels shall be functional.

F. No control rod drive pump shall be in service.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Control rod position shall be verified in accordance with Table 4.2.A.

B. Each directional control valve shall be verified to be electrically disarmed at intervals not to exceed once every 3 days.

C. Each control rod insert drive valve shall be verified closed at intervals not to exceed once every 3 days.

D. The accumulator pressure shall be checked once a day.

E. The count rate shall be recorded once each shift.

F. The control rod drive pump discharge pressure shall be checked once each shift.

3.3/4.3 BASES: REACTIVITY CONTROL

To prevent an inadvertent or spurious withdrawal of a control rod the directional control valves of each control rod have been electrically disarmed. As a further precaution, the valve in the drive water supply to each rod will be closed. In the unlikely event that a control rod does become withdrawn, two channels of the SRM's are required to be available for visual indication of neutron level. Although the SRM's may not immediately respond to a single rod movement, they would be adequate to monitor the approach to criticality. Also the SRM's are connected in the non-coincidence scram mode to provide rapid rod insertion from a high-high count rate on either SRM. Additionally, a manual scram will also be available. To assure that the control rods can be scrammed, the control rod accumulators are required to be charged with nitrogen and water pressure and a control rod drive pump is required to be in service. To provide additional indication of a control rod withdrawal, the control rod position indicator full-in switches will be functional for every control rod and will provide indication in the control room or the control rod position will be verified to be full-in by a continuity check of the full-in limit switch at the local panel.

3.4/4.4 STANDBY LIQUID CONTROL SYSTEM

Delete present section in its entirety (pages 122-129) and replace with the following:

LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. The standby liquid control system shall be functional at all times.

2. The injection line manual isolation valve 63-524 will be closed except when the system is required for emergency injection.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested.

2. At least once during each operating cycle:

a. Check that the setting of the system relief valves is 1425 ± 75 psig.

b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm

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LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.4 STANDBY LIQUID CONTROL SYSTEM	4.4 STANDBY LIQUID CONTROL SYSTEM
	against a system head of 1275 psig shall be verified. After pumping boron solution, the system shall be flushed with demineralized water.
	c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel. This test checks explosion of the charge associated with the tested loop, proper operation of the valves and pump operability. Replacement charges shall be selected such that the age of charge in service shall not exceed five years from the manufacturers assembly date.
	d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.
B. Operation with Inoperable Components:	B. Surveillance with Inoperable Components:
1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled. The inoperable component shall be returned to service within seven days.	1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

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LIMITING CONDITIONS FOR OPERATION

3.4 STANDBY LIQUID CONTROL SYSTEM

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

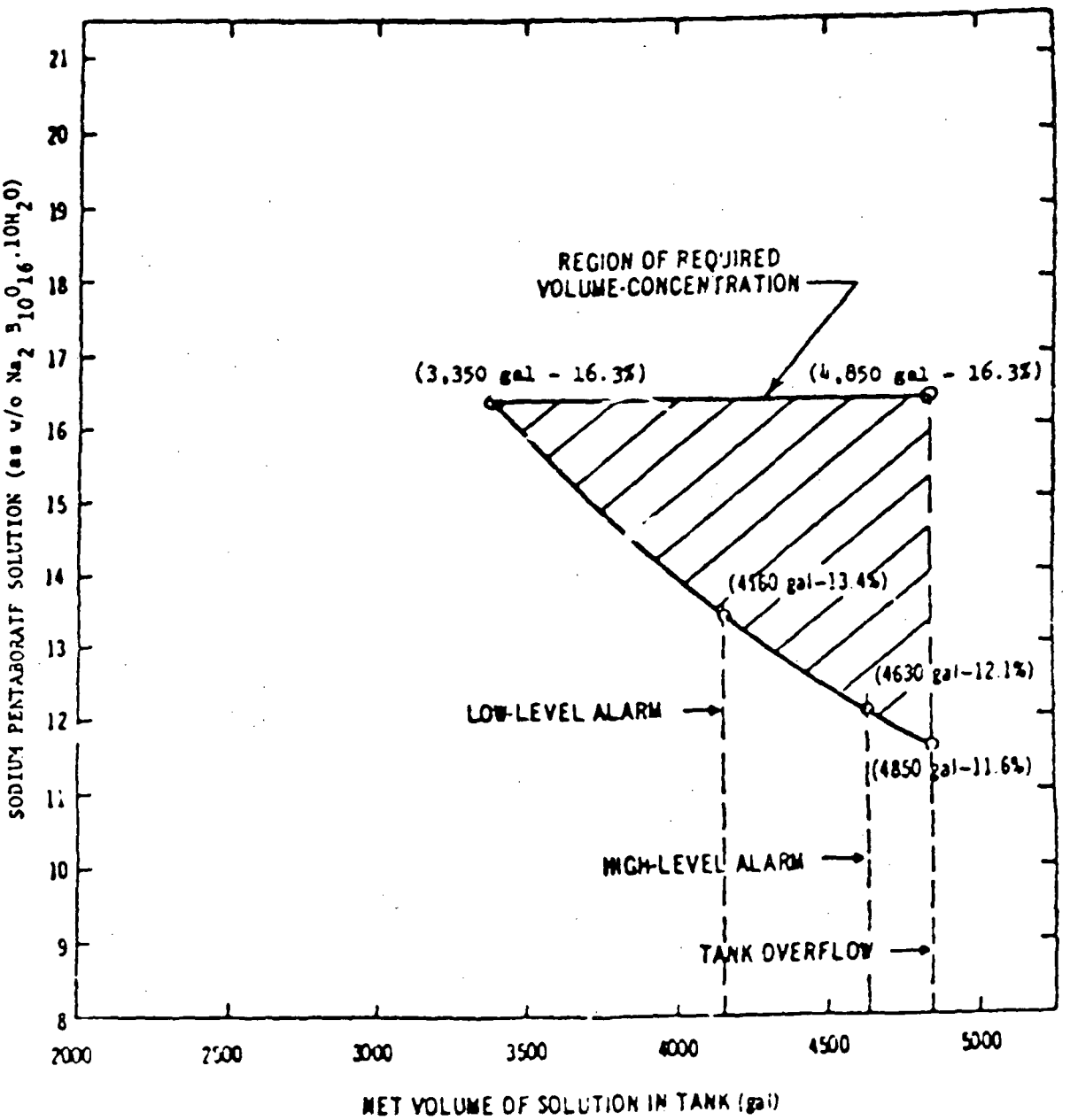
C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

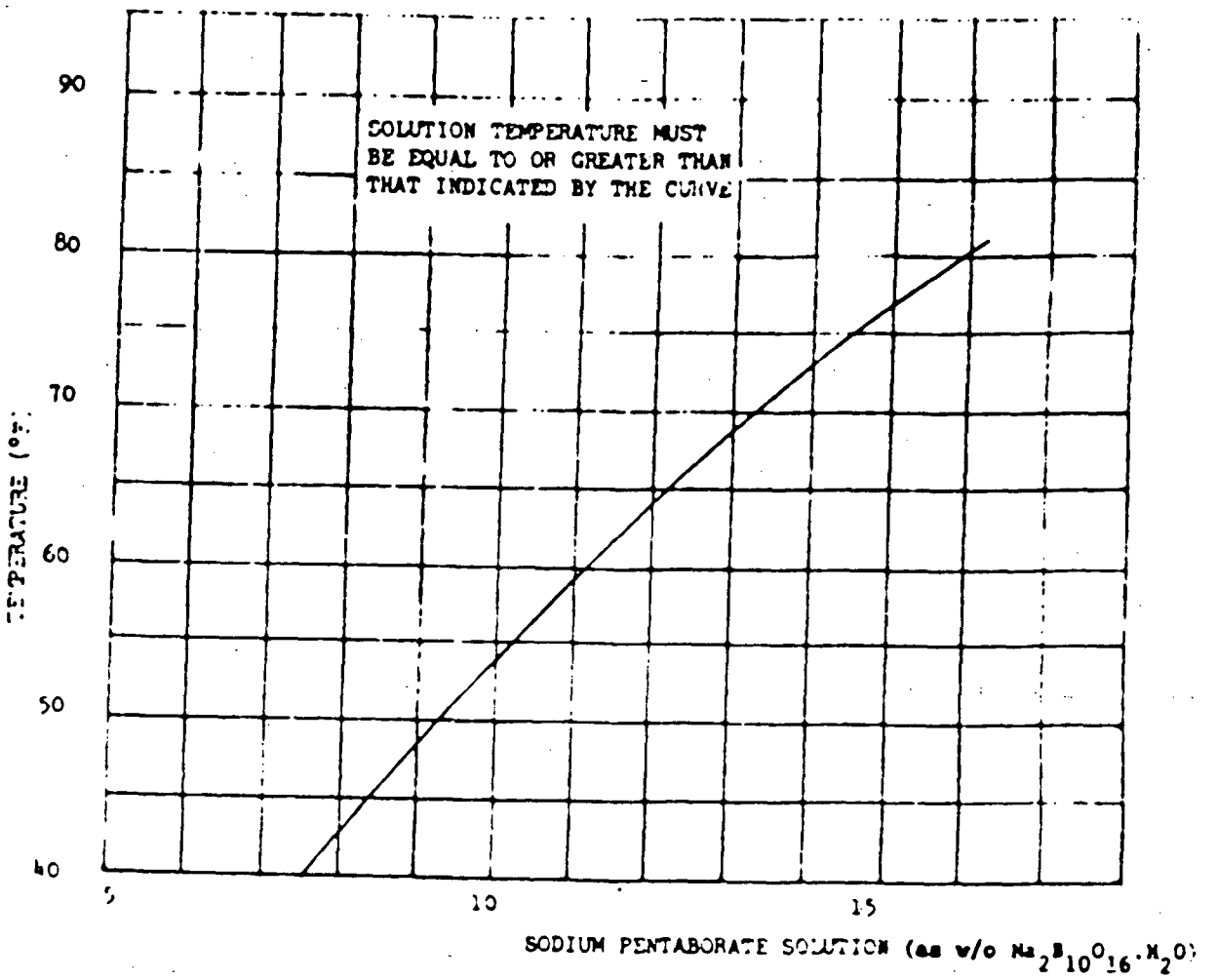
3. Concentration: Check at least once per month. Also check concentration any time water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
VOLUME-CONCENTRATED REQUIREMENTS

FIGURE 3.4-1



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS

FIGURE 3.4-2

BASES: STANDBY LIQUID CONTROL SYSTEM (Continued)

C. (Continued)

The volume concentration requirement of the solution are such that should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 4,160 gallons of solution having a 13.4 percent sodium pentaborate concentration or the equivalent is required to meet this shutdown requirement as defined in Figure 3.4.1.

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

BASES: STANDBY LIQUID CONTROL SYSTEM

A. The purpose of the liquid control system is to provide the capability of maintaining the reactor in a cold shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. One inoperable pumping circuit does not immediately threaten shutdown capability. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained from a one-out-of-two system.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10° F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

3.5/4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Delete the present specification in its entirety (pages 130-150b) and insert the following:

LIMITING CONDITIONS FOR OPERATION

1.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and containment cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. When the reactor vessel is atmospheric and irradiated fuel is in the reactor vessel, at least one core spray loop with one pump and available diesel generator shall be capable of delivering flow.

B. Residual Heat Removal System (RHRS) Containment and Shutdown Cooling

1. When the reactor vessel

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core and suppression pool cooling systems when the corresponding limiting conditions for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing

Item	Frequency
a. Pump operability	upon restoration and monthly thereafter
b. Motor operated valve operability	upon restoration and monthly thereafter

2. When it is determined that one core spray loop with one pump and available diesel generator is incapable of delivering

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

is atmospheric and irradiated fuel is in the reactor vessel, at least one RHR loop with two pumps or two loops with one pump per loop shall be capable of manual operation. Diesel generators must also be available to power the pumps. A service water supply must be available.

C. RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

1. When the reactor vessel is atmospheric and irradiated fuel is in the reactor vessel, 4 (or 3) RHRSW pumps must be functional and aligned to RHR header service corresponding to the 4 selected RHR pumps (2 of which may have corresponding heat exchangers on one RHRSW header thus requiring only 3 RHRSW pumps).

2. At all times, 2 RHRSW pumps which are capable of automatic restarting shall be

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

flow, another core spray pump with an available diesel generator shall be selected, and all active components in the flow paths shall be immediately demonstrated to be capable of delivering flow.

B. Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

1. Residual Heat Removal System Testing

	Item	Frequency
a.	Pump operability	Upon restoration and monthly thereafter
b.	Motor operated valve operability	Upon restoration and monthly thereafter

2. When it is determined that one RHR pump (containment and suppression pool cooling) or associated heat exchanger is incapable of delivering flow and removing heat at a time when flow capability and heat removal is required, the remaining RHR pump and associated heat exchanger and available diesel generator, and all active components in the flow paths shall be demonstrated

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

assigned to EECW header service with one pump assigned to each header and running continuously. Each pump shall be assigned to a separate diesel power supply.

D. Maintenance of Filled Discharge Pipe

Whenever the core spray system or RHR systems are required to be functional, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The condensate head tank shall be aligned to serve the discharge piping of the RHR and CS pumps. The pressure indicators on the discharge piping of the RHR and CS pumps shall indicate not less than listed below.

PI-75-20	70 psig
PI-75-48	70 psig
PI-74-51	70 psig
PI-74-65	70 psig

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

to be capable of delivering flow and heat removal immediately and weekly thereafter until the inoperable RHR pump and associated heat exchanger is returned to service or an alternate pump and heat exchanger with an available diesel generator selected and verified.

C. RHR Service Water System (RHRSWS) And Emergency Equipment Cooling Water System (EECWS)

1. RHR Service Water System

Each of the 4 (or 3) RHRSW pumps and associated essential control valves on the RHR heat exchanger headers shall be demonstrated to be functional upon restoration and once every three months thereafter if not in continuous service.

2. EECW System

Each of the 2 RHRSW pumps assigned to EECW service will be in continuous service. Associated essential control valves on the EECW headers shall be demonstrated to function once every three months.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

3. When it is determined that one RHRSW pump and associated control valves on an RHR heat exchanger header is incapable of delivering flow at a time when flow delivery capability is required, the remaining RHRSW pumps and associated heat exchangers and available diesel generators, and all active components in the flow paths shall be demonstrated to be capable of delivering flow immediately and weekly thereafter until the inoperable RHRSW pump is returned to service or an alternate pump is selected and verified.

4. When it is determined that one RHRSW pump and associated control valves on an EECW header is incapable of delivering flow at a time when flow delivery capability is required, an alternate RHRSW pump on a corresponding diesel generator shall be selected and assigned to the same EECW header.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.5 CORE AND CONTAINMENT COOLING SYSTEMS	4.5 CORE AND CONTAINMENT COOLING SYSTEMS

B. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray system and RHR system are filled:

1. Prior to the operation of the RHRS and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined.
2. Following any period where the RHR or core spray systems have not been required to be operable, the discharge piping of the nonfunctional system shall be vented from the high point prior to the return of the system to service.
3. When the RHRS and CSS are required to be functional the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure logged.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.A Core Spray System (CSS)

The core spray loop provides one of several means of supplying water to the reactor vessel. The capacity of one core spray pump is in excess of 10 times the amount of water required to cool the reactor core in its present shutdown condition. Power operation of the reactor will not be permitted, therefore, the margin available for this purpose will increase with time. Technical Specification 3.5.A and 3.5.B require that at least three pumps will be available to assure that adequate cooling of the fuel is accomplished. However, considerably more equipment is available, including; additional core spray, RHR, control rod drive, condensate booster, hotwell and raw water pumps. As noted in the introduction availability of these components will be maximized.

Analyses have been performed to demonstrate that in excess of 15 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that a core spray loop could be manually placed in operation in less than one hour. Hence considerable margin is available for manual actuation.

3.5 BASES: CORE CONTAINMENT COOLING SYSTEMS

3.5.B Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling

The decay heat removal requirements for one unit in the cold shutdown condition can be conservatively met by the operation of one RHR pump and its associated RHR heat exchanger in the shutdown cooling mode. The total heat load for the heat exchanger is estimated to be less than one-fourth of the heat exchanger capability under the required flow and temperature conditions. At least three RHR pumps and associated heat exchangers on each unit are capable of delivering flow and removing the decay heat. The low decay heat and absence of pressure which could foster an unacceptable loss of coolant allows ample time for manual operation in accordance with established operating instructions.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.C RHR Service Water System (RHRSW) Emergency Equipment Cooling Water System (EECWS)

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump on one heat exchanger on each unit. One RHRSW pump is required foreach unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allows ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps assigned to EECW service are running continuously with each assigned to a seperate supply header. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are arranged with automatic restart provisions in the event that board undervoltage should occur.

3.5 BASES: CORE AND CONTAINMENT COOLING SYSTEMS

3.5.D Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray and RHR system are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in a functional condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be nonfunctional for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow prior to any pump operation to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a head tank located approximately 100 feet above the discharge line high point supplies makeup water for these systems. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and approximately 70 psig for a water level at the head tank and are monitored daily to ensure that the discharge lines are filled.

3.6/4.6 PRIMARY SYSTEM BOUNDARY

Delete the present specification in its entirety (pages 151-175) and substitute the following:

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the status of the reactor coolant system.

Objective

To assure the integrity of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.

B. Coolant Chemistry

Reactor water shall be maintained within the following limits:

Conductivity,
10 umhos/cm@25°C
Chloride, 0.5 ppm

C. Safety and Relief Valves

Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250v DC MOV boards.

D. Reactor Vessel Head Vent

The reactor vessel head vents shall be in the open position.

SURVEILLANCE REQUIREMENT

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitations

When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

B. Coolant Chemistry

1. A sample of reactor coolant shall be analyzed:

   a. at least every 96 hours for conductivity and chloride ion content.

   b. at least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor is inoperable.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
1.6 PRIMARY SYSTEM BOUNDARY	4.6 PRIMARY SYSTEM BOUNDARY C. Safety and Relief Valves The position indicating lights for the four safety relief valves specified in 3.6.C.1 shall be checked each operating shift to confirm control power available for actuation. D. Reactor Vessel Head Vent The position indicating lights on the reactor vessel head vents shall be checked each operating shift to confirm the valves are indicated to be in the open position.

4.6.A/4.6.A BASES: THERMAL AND PRESSURIZATION LIMITATIONS

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The reactor vessel head flange and head are constructed such that their initial maximum NDTT is 10° F, as cited in paragraph 4.2.7 of the safety analysis report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 40° F + 60° F or 100°F.

4.6.B/4.6.B BASES: COOLANT CHEMISTRY

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited becasue it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations and dissolved oxygen. Zircaloy does not exhibit similar stress corrosion failures.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. When conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values becaues conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's however, where no additives

BASES: COOLANT CHEMISTRY (Continued)

are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded.

Methods available for correcting the off-standard conditions, include using the cleanup demineralizer, reducing the input rate of impurities, or changing the supply of cooling water. The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

BASES: SAFETY AND RELIEF VALVES

With the reactor in the cold shutdown condition and with low decay heat, four safety relief valves are more than adequate to assure the integrity of the reactor coolant systems. Control power from different 250V MCV boards assures availability of two valves even in the unlikely event of loss of a 250V MCV board. Surveillance checks each shift assures control power is available for valve actuation.

BASES: REACTOR VESSEL HEAD VENT

Maintaining the reactor vessel head vents in the open position with the RHR shutdown cooling system in service keeps the reactor vessel at atmospheric pressure when in the cold shutdown condition. Surveillance checks of the position indicating lights are made each shift.

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3.7/4.7 CONTAINMENT SYSTEMS

Delete the present specification in its entirety (pages 175 throught 216)

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
Applicability	Applicability
Applies to the status of the primary containment system.	Applies to the primary containment.
Objective	Objective
To monitor pressure and temperature of the primary containment system.	To verify the pressure and temperature of the primary containment.
Specification	Specification
A. The suppression chamber water level and temperature shall be monitored (see Table 3.2.A).	A. The suppression chamber water level and temperature shall be checked once per day.
B. The drywell pressure and temperature shall be monitored (see Table 3.2.A).	B. The drywell pressure and temperature shall be checked once per day.

3.7 BASES: CONTAINMENT SYSTEMS

This specification ensures indication of adequate information regarding status of the drywell pressure and temperature and suppression chamber water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

3.8/4.8 RADIOACTIVE MATERIALS

Pages 217-220 are unchanged. Page 221 - replace with the attached page which revises specification 3.8.B.8. Page 222 is unchanged.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
8. Radioactive gases released from each units turbine and reactor building roof vents, the radwaste building roof vents, and the main stack shall be continuously monitored. To accomplish this, at least one reactor building and one turbine building vent monitoring system per unit shall be operating whenever that unit's building ventilation system is in service. Also one radwaste building system vent monitoring channel shall be operating whenever the radwaste building ventilation system is in service. At least one main stack monitoring channel shall be operating whenever any units air ejector, mechanical vacuum pump, or a standby gas treatment system train is in service. If normal monitoring systems are not available, temporary monitors or other systems shall be used to monitor effluent. A monitoring channel may be out of service for 4-hours for functional testing and calibration without providing a temporary monitor.	4.8.B Airborne Effluents
9. The primary containment shall be purged through the standby gas treatment system.	C. Radiological Environmental Monitoring Program An environmental monitoring program shall be conducted as described below and outlined in Table 4.8.F. 1. Atmospheric Monitoring a. The atmospheric monitoring network is divided into three subgroups consisting of 12 monitoring stations of which 9 stations shall be operable at all times. The monitoring stations are shown on Figure 4.8-1. These monitoring locations are subject to change dependent upon continued evaluation of the environmental monitoring program. The station at Muscle Shoals will be used as background reference. Each monitor shall be capable of continuously sampling air at regulated flow of approximately three cubic feet per

3.8/4.8 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

There is no change to thi specification (pages 223 - 239).

3.9/4.9 Delete the present specification in its entirety (pages 240-248) and substitute the following:

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. 3 diesel generators available and their associated 4 kV shutdown boards. All 4 kV shutdown boards energized unless offsite power is lost.
3. 4 kV shutdown bus 1 or 2 energized.
4. 480V shutdown boards associated with each unit are energized.
5. Under voltage relays on the three 4 kV shutdown boards associated with the functional diesel generators shall be available.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater.

During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. 3 diesel generators available and their associated 4 kV shutdown boards.
3. [illegible] kV shutdown boards energized unless offsite power is lost.
4. 4 kV shutdown bus 1 or 2 energized.
5. 480V shutdown bus associated with [illegible] energized.
6. [illegible] on the three 4 kV [illegible] associated with [illegible] diesel generators [illegible] [illegible]

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load [illegible]

During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be verified and the diesel starting time to reach rated voltage and speed shall be logged.

4.9 AUXILIARY ELECTRICAL SYSTEM

b. Once a month the quantity of diesel fuel available shall be logged.

c. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

6. 250V DC shutdown board batteries and chargers for the three functional boards must be available.

7. The 250V unit board batteries and a battery charger for each battery and associated battery boards must be functional.

8. Batteries, chargers, busses, and feeds to the four SRM channels specified in Table 3.2.A shall be operable in each unit.

9. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.

B. Cold Shutdown on Units 1 And 2 With Inoperable Equipment

1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.

2. When one of the three diesel generators become inoperable and a shutdown bus and 3-4 kV shutdown boards are available, 2 days are allowed

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

d. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.

2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) And Shutdown Board Batteries (250-Volt).

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and over-all battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

3. Undervoltage Relays

a. Once every 6 months, the

LIMITING CONDITIONS FOR OPERATION

AUXILIARY ELECTRICAL SYSTEM

to comply with 3.9.A.2

3. When one 4-kV shutdown board is inoperable, both off-site 161-kV transmission lines and both common station transformers shall be available and the remaining 4-kV shutdown boards and associated diesel generators, and all 480V emergency shutdown boards shall be functional.

4. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

5. From and after the date that one of the three 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

b. The undervoltage relays which start the diesel generators from the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.

B. Operation with Inoperable Equipment

1. When one common station transformer, or one 161 kV line is found to be inoperable, all diesel generators and associated boards must be demonstrated to be functional immediately and daily thereafter.

2. When one diesel generator is found to be inoperable, the remaining diesel generators and associated boards shall be demonstrated to be functional immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM plans to return the failed component to an operable state.	4.9 AUXILIARY ELECTRICAL SYSTEM 3. When one 4-kV shutdown board is found to be inoperable, all remaining 4-kV shutdown boards and two associated diesel generators, shall be demonstrated to be functional, immediately and daily thereafter.

3.9 BASES: AUXILIARY POWER SYSTEM

The auxiliary power system consists of 3 functionally operable diesel generators (A, B, & D) capable of automatic start on loss of voltage to their respective 4-kV shutdown boards. The C diesel feed is inoperable but its associated 4-kV shutdown board is available. The 480V shutdown boards 1A, 1B, 2A, and 2B are operable with at least one source of power. All the 480V reactor MOV boards 1A, 1B, 1C, 2A, 2B, & 2C are operable with at least one source of power.

The requirements of this specification are based on the minimum complement of equipment to redundantly maintain the reactor in a safe condition. The capacity requirements of this mode of operation can be adequately addressed with two diesels. The third diesel is to accomodate loss of off-site power and single failures in the diesel generator plant.

All 4-kV and 480V boards are available and the shutdown busses are available. With a single failure in any of the 4-kV and 480V board or feeders, diesel power can be directed to an adequate complement of equipment to accomodate the shutdown mode. This is accomplished by either feeding the required boards and their associated loads directly or by feeding two boards from one diesel through the backfeed capability of the shutdown bus.

There are eight 250-volt d-c battery systems each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4160-volt shutdown boards.

3.9 BASES: AUXILIARY POWER SYSTEM (Continued)

Each 250-volt d-c shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the stand-by diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe cooldown of both units in the event of the loss of offsite power and single failure in any one unit. The loss of one 250-volt shutdown board battery affects normal control power only for the 4160-volt shutdown board which it supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

4.9 BASES

The monthly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The diesels will be loaded to at least 75 percent of rated power while engine and generator temperatures are stabilized (about one hour). The minimum 75 percent load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no overheat problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and to correct any mechanical or electrical deficiency before it can result in a system failure.

Battery maintenance with regard to the floating charge, equilizing charge, and electrolyte level will be based on the manufacturer's instruction and sound maintenance practices. In addition, written records will be maintained of the battery performance. The plant batteries will deteriorate with time but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the Ampere-hour capacity of the battery, and will be applied as recommended.

3.10/4.10 CORE ALTERATIONS

The present technical specification (pages 249 - 260) remains unchanged.

3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS

Add a new specification as indicated on the attached sheet, p. 260a.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS Applicability Applies to construction activities of unit 3. Objective To assure that unit 3 construction activities do not adversely affect the safety of either unit 1 or 2. Specification All operations, tests, and other activities associated with the construction of unit 3 that can affect the safety of either unit 1 or unit 2 are prohibited.	

3.11 BASES: UNIT 3/UNIT 1 AND 2 INTERACTIONS

Interface documentation that presents the results of analyses to demonstrate that adverse affects associated with interaction between unit 3 and either unit 1 or unit 2 are not available. Since the arrangement of safety systems is such that adverse affects are possible, further degradation in either unit 1 or unit 2 should not be allowed until the interaction analysis is completed.

5.0 MAJOR DESIGN FEATURES

The present technical specification remains unchanged.

6.0 ADMINISTRATIVE CONTROLS

The present specification remains unchanged (pages 263 - 288) except delete page 285 and substitute the following pages which contain a new table 6.8.A and its bases.

TABLE 6.8.A

Minimum Shift Crew Requirements

Shift Position		Type of License
Shift Engineers (SE)	1	SRO
Assistant Shift Engineers (ASE)	1	SRO
Licensed Reactor Operator[1]	1	RO
Unit Operators (UO)	2	RO
Assistant Unit Operators (AUO)	6	None
Health Physics Technician	1	None
Other Personnel	2	
Minimum Shift Crew	14	

Notes: SRO - Senior Reactor Operator
RO - Reactor Operator

Note for Table 6.8.A

[1]This position is normally filled by an assistant shift engineer, but as a minimum it may be filled by a licensed reactor operator. When the incumbent is not a senior reactor operator, he shall not be assigned duties requiring him to direct licensed activities of reactor operators.

BASES

As a result of the damage created by the fire to various units 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located throughout the facility. Analyses have shown that in excess of 15 hours are available to perform these various tasks. In addition, studies have been performed to assess the amount of time

BASES (Continued)

required to perform these manual operations including consideration of additional equipment failures. Results of these studies have shown that operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours. These studies were performed with the assumption of a shift crew of 10 men. The additional 4 men required provide margin as well as to allow for continued monitoring during performance of these manual actions. It is expected that two of these four men would be used for that purpose. When the equipment is aligned as described in Appendix A to "Safety Analysis of the Browns Ferry Nuclear Plants Units 1 and 2 During Operations Related to Removal of Damaged Cables, Cable Trays, and Conduits," dated May 1, 1975, the minimum crew size could be reduced.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 9 TO FACILITY LICENSE NO. DPR-33

AND AMENDMENT NO. 6 TO FACILITY LICENSE NO. DPR-52

(CHANGE NO. 10 TO TECHNICAL SPECIFICATIONS)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-259 AND 50-260

INTRODUCTION

On March 22, 1975, a fire at the Browns Ferry facility resulted in substantial damage to power, control, and instrumentation wiring. The facility is not operational and is in a cold shutdown condition with fuel in the core and the vessel vented to the containment atmosphere. Although many systems were damaged, the facility has been maintained in a safe and stable condition utilizing a substantial number of systems and components. In fact, the functioning systems exceed the minimum number usually required for a facility in cold shutdown condition.

In order to prepare the facility for repair of the fire damage, vital equipment to protect the core from damage is being reconnected, or rerouted or verified to be free from the fire affected wiring or systems. Many of these vital systems are being modified to manual control rather than automatic control to isolate them from spurious operation resulting from interconnection with fire damaged wiring or systems. Some systems or components are being disconnected or locked in a safe configuration. The primary system mode switch is, by administrative control, locked in the "shutdown" mode.

Before any repair and removal of cables and systems from the fire affected zone can be authorized, all of the necessary vital systems must be protected, isolated, or verified as free from interconnection with any of the systems or cabling which is to be removed. These activities of rerouting, reconnecting, verification, and isolating systems from potential interaction with systems or wiring to be removed are presently in progress.

The available systems and components to provide plant safety under present conditions, while the facility is in cold shutdown and the plant is being configured into a condition protected against possible adverse interaction resulting from repairs and restoration, have been reviewed and appropriate technical specifications governing this period have been established through conferences between the staff and the licensee and submitted for approval by the licensee. This safety evaluation describes the protection required to provide adequate safety during present conditions. It does not address or authorize cable removal, cutting, or repair of the fire affected zone. Any such authorization will be addressed in a subsequent safety evaluation.

EVALUATION

1. Nuclear Safety

The control rods are each fully inserted in the core. The reactor mode switch is locked in the shutdown position, the control rod directional control valves, which provide the drive fluid to withdraw the rods, have been electrically disconnected, and the manual valves in the drive water supply lines have been closed. In this condition the core is subcritical and cannot be made critical. Under the present low pressure, low temperature conditions, there are no equipment malfunctions or single operator errors which could cause any control rod to withdraw. Nonetheless, rod position is to be verified in the fully-in position on a regular basis. The control rod position indicator full-in switches will be operable or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel.

In addition, the electrically disconnected condition of the control rod directional control valves will be verified by visual observation, along with visual observation of the closed position of the manual valve in the drive water supply lines.

In addition to the manual scram, an automatic alarm and scram function has been added for the operable source range monitors, even through this is not ordinarily a function of this system. This system will detect any approach to criticality and will both alarm and automatically cause scram action to drive the

control rods in if for any unforeseen reason they might have moved outward. The control rod accumulators will be maintained in a fully charged condition. A standby supply of borated water is available for injection into the reactor to further ensure a subcritical core. These systems provide a high degree of assurance that the facility is protected against the occurrence of criticality under present conditions.

2. Core Cooling and Water Inventory

Under present conditions, the fuel remains in the core under cold shutdown conditions (less than 212 F and atmospheric pressure). Core cooling is presently being provided by the operation of one residual heat removal (RHR) pump and associated heat exchanger. Decay heat which must be removed to prevent fuel damage is very small, amounting to less than one-fourth of the heat exchanger capacity of one RHR system. Nonetheless, two RHR pumps and heat exchangers are required to be operational by the technical specifications and at least three RHR pumps and heat exchangers are presently available on each unit. The low decay heat level provides ample time for manual operation of backup equipment. The RHR pumps take suction from a recirculation loop and pump water through the RHR heat exchangers in which reactor decay heat is transferred to a separate RHR service water system. In the event that the RHR pump providing flow should fail, the second RHR system for the unit is functioning and can provide ample cooling. In the event of the failure of the single valve in the suction line, heat removal can be accomplished by the generation of steam in the core which is vented to the containment through the open vent valves or backup relief valves. The condensed steam returns to the torus which can be cooled by either of the RHR systems operating in a torus cooling mode which does not depend on the failed suction valve. Makeup coolant to the core in this mode of operation could be provided by either of the RHR pumps or the core spray pump taking suction from the torus.

Analyses have been performed to demonstrate that in excess of 15 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that backup systems could be manually placed in operation in less than 1 hour. Considerable margin is, therefore, available for manual actuation.

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump or one heat exchanger on each unit. One RHRSW pump is required for each unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allows ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps assigned to EECW service are running continuously with each assigned to a separate supply header. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are arranged with automatic restart provisions in the event that board undervoltage should occur.

In the present cold shutdown vented condition, with no pressure and low temperature in the primary system, there is no source to cause piping damage. Loss of coolant through inadvertent draining is minimized by locking valve positions so that water cannot be drained from the vessel. However, even in the event of loss of coolant inventory either of the RHR pumps or the core spray pump are capable of taking suction on the torus and providing water to the vessel. Even beyond these pumps there are additional RHR pumps, core spray pumps, and even hotwell and condensate booster

pumps capable of supplying coolant to the vessel. Although the technical specifications do not require that these pumps be kept functioning, the technical specifications do provide that the licensee will not deliberately remove any of the pumps or any other component capable of achieving the objectives of the technical specifications from service except for maintenance required to prevent a substantial increase in the risk of failures to the component or other functioning system.

3. Electrical Power

The electrical energy to operate the systems required to maintain adequate cooling of the core is provided by redundant offsite and onsite power supplies. The offsite power is available through two 161 kV circuits connected from the power grid to redundant shutdown buses which in turn supply power to all four 4.16 kV emergency buses. To provide power in the event of loss of the offsite power supply, three of the four plant diesels are functioning along with their associated emergency buses. Only two of the three are required to supply a sufficient amount of electrical energy to power vital equipment. The emergency diesels are cooled through the emergency coolant water system which is presently continuously operating and has been configured to automatically transfer to diesel load in the event of loss of offsite power to assure that these pumps automatically provide sufficient coolant to the diesels to protect against overheating. Redundant 250 V batteries and their associated distribution systems are available to provide DC control power as necessary to the required system.

4. Minimum Shift Crew Considerations

As a result of the damage created by the fire to various Unit 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located through the facility. Analyses have shown that in excess of 15 hours are available to perform various tasks involved in transferring to the various functioning backup systems. Manual operation of equipment for which the most degradation is experienced could be achieved in

a time interval of several hours, and the licensee has indicated that these operations could easily be accomplished by a shift crew of 10 men. We have required a shift crew of 14 men. The additional four men required provide margin, as well as to allow for continued monitoring during performance of these manual actions. It is expected that two of these four men would be used for that purpose.

5. Quality Assurance

In connection with the work of configuring the plant so various systems and components are free of potential interaction during fire damage removal and facility restoration, the QA procedures have recently been clarified to focus on this work to assure that they conform to the approved QA program. The previously approved QA program itself has not been modified.

6. Unit 3 Construction Activities

As a result of interfaces between Unit 3 and Unit 1 and 2, the technical specifications do not permit operations, tests, and other activities associated with the construction of Unit 3 that could affect the safety of either Unit 1 or 2.

7. Summary

In its present cold shutdown condition, the facility has been protected from damage to the core by disarming control rods so that no mechanism is available which could result in core criticality. The available functioning equipment is adequate to provide ample coolant flow to keep the core cooled and exceeds the minimum required for cold shutdown by the technical specifications governing facility operation prior to the fire.

The technical specification changes associated with this amendment reflect the changes that are necessary to account for the present condition of the plant. Certain additional controls and equipment requirements not required in the pre-fire technical specifications have been added to provide additional assurance that the plant will be maintained in a safe and stable shutdown

condition during the present activities of preparing for the restoration program. The technical specifications associated with this amendment include these added controls and equipment requirements.

Except as necessitated by the physical realities that exist due to fire damage, no safety limit, limiting condition for operation, or surveillance requirement in the pre-fire technical specifications that is pertinent to the present cold shutdown condition of the plant has been modified, relaxed, or deleted by this amendment. The resulting technical specifications continue to provide adequate protection to the health and safety of the public and, by accounting for the actual existing condition of the plant systems and components, they minimize potentially unsafe conditions or actions. With the plant locked into a shutdown condition and considering the very low amount of decay heat generation from the shutdown core, the probability of any accident involving the release of radioactivity from the site is very low and the potential consequences of any accident are correspondingly low. In view of the foregoing, we conclude that the issuance of these amendments does not involve a significant hazards consideration.

CONCLUSION

The changes in technical specifications authorized in connection with this evaluation result in enhancement of safety under present conditions, as discussed above. Based on these considerations, we have concluded that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 9, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 9 to Facility Operating License No. DPR-33 and Amendment No. 6 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority which reivsed Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to assure that the two units will remain in a safe and stable posture during the period of plant shutdown resulting from damage due to a fire which occurred on March 22, 1975. The amendments do not authorize removal of fire damaged components or systems or restoration of the facility to operable conditions. This will be subsequently considered.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendments, Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated May 8, 1975, (2) Amendment No. 9 to License No. DPR-33 and Amendment No. 6 to License No. DPR-52 with Change No. 10, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 9th day of May, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT III

LETTER FROM THE NRC TO THE TVA DATED JUNE 13, 1975, WITH ENCLOSURES

AMENDMENT NO. 10 TO LICENSE NO. DPR-33

AMENDMENT NO. 7 TO LICENSE NO. DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 13 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

The Commission has issued the enclosed Amendment No. 10 to Facility License No. DPR-33 and Amendment No. 7 to Facility License No. DPR-52 for the Browns Ferry Nuclear Plant, Units 1 and 2. These amendments include Change No. 11 to the respective Technical Specifications and are in response to your request dated June 2, 1975. Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the two units will remain in a safe and stable posture during the period of defueling and fuel storage in the fuel pools. The plants will be maintained in this condition until completion of the repairs of damage resulting from the fire which occurred on March 22, 1975. We have completed our review of your proposed plans for removal of fire-affected features and find them acceptable. Accordingly, pursuant to Mr. Rusche's letter to you dated April 17, 1975, these amendments authorize you to commence removal of fire-affected features upon fulfillment of the requirements of Technical Specification 3.10.H.

Our review of your proposed plans related to restoration of Units 1 and 2 is continuing and will be the subject of a subsequent approval action.

Sincerely,

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
See next page

Enclosures:
1. Amendment No. 10 to License No. DPR-33
2. Amendment No. 7 to License No. DPR-52
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
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Athens Public Library
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Mr. Thomas Lee Hammons
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State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, NE
Atlanta, Georgia 30309

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 2, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-33 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 11."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 11 to Technical Specifications

Date of Issuance: JUN 13 1975

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated June 2, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-52 is hereby amended to read as follows:

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 11."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 11 to Technical Specifications

Date of Issuance: JUN 13 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 10 TO LICENSE NO. DPR-33

AND LICENSE AMENDMENT NO. 7 TO LICENSE NO. DPR-52

(CHANGE NO. 11 TO THE TECHNICAL SPECIFICATIONS)

DOCKET NOS. 50-259 AND 50-260

These page changes refer to pages of the "Technical Specifications and Bases for Browns Ferry Nuclear Plant, Units 1 and 2, effective prior to May 9, 1975

Revise Appendix A as follows:

Remove Pages	Insert New Pages
8 through 26	8, 15
27 through 30	27, 28
31 through 49	31, 33, 42
50 through 106	50, 54, 59, 81, 99, 100
107 through 121	108, 109, 115
122 through 129	122, 123, 124, 125, 126, 127, 128
130 through 150b	130, 131, 131a, 132, 133, 134, 135, 136, 143, 144, 145, 146, 147, 148, 149
151 through 174	151, 152, 166, 167, 168
175 through 216	175, 176, 176a, 206
221	221
240 through 260	240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 250, 251, 252, 253, 254, 254a, 255, 256, 256a, 257, 258, 259, 260, 261, 262, 263, 264
--	270A, 270B
285	285, 285a

FACILITY OPERATING LICENSES DPR-33 AND DPR-52

INTERIM TECHNICAL SPECIFICATIONS AND BASES

FOR

BROWNS FERRY NUCLEAR PLANT

UNIT 1 AND UNIT 2

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior.

Objective

To establish limits which ensure the integrity of the fuel cladding.

Specification

A. Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 inches above the top of the normal active fuel zone.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the level of the process variables at which protective action is initiated to prevent the fuel cladding integrity safety limit from being exceeded.

Specification

The limiting safety system settings shall be as specified below:

A. Manual operation under procedural controls shall be established to assure the reactor water level remains $\geq$378 inches above vessel zero.

1.1/2.1 BASES: LIMITING SAFETY SYSTEM SETTINGS RELATED TO FUEL CLADDING INTEGRITY

The objective of this specification is to assure that irradiated fuel in the reactor vessel remains covered with water at all times. When plant equipment as specified in facility technical specification is available, this objective is accomplished automatically. This is no longer possible. Accordingly, procedural control of the water level is required as set forth in the specification 3.2.A.1.

With the reactor in cold shutdown, rapid makeup of coolant inventory is not required. The provisions for manual operation under procedural controls provided by the TVA Safety Analysis of the BFNP units 1 and 2 included as Part VI, Section E, of the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" will assure timely isolation of leaks by requiring redundant level alarms and indicators. Double isolation features, one of which shall be operative from the control room, will prevent draining of the vessel as a result of any single active component failure. Also, in the unlikely event that coolant inventory is required, a core spray pump, capable of delivering flow, will be available to provide the required vessel makeup capability.

The limiting safety systems scram setpoints are not capable of performing their intended function since the automatic actions they would initiate are unavailable. However, the facility has been placed in a condition that prevents the need for these actions.

SAFETY LIMIT

1.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to limits on reactor coolant system pressure.

Objective

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification

A. The pressure at the lowest point of the reactor vessel shall not exceed 1,375 psig whenever irradiated fuel is in the reactor vessel.

LIMITING SAFETY SYSTEM SETTING

2.2 REACTOR COOLANT SYSTEM INTEGRITY

Applicability

Applies to the devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective

To define the protective action to prevent the pressure safety limit from being exceeded.

Specification

A. The reactor vessel head vent valves shall be kept in the open position.

B. Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards, whenever the reactor vessel head is bolted in place.

1.2/2.2 BASES: REACTOR COOLANT SYSTEM INTEGRITY

With the reactor in the cold shutdown condition and with redundant RHR shutdown cooling capability, pressurization of the reactor vessel is prevented. In the unlikely event that shutdown cooling capability from the RHR system is lost, manual operation of any one of the four available relief valves prevents pressurization of the reactor pressure vessel above 50 psig preserving the safety limit whenever the reactor vessel head is bolted in place. This keeps the pressure low enough to permit addition of water by a core spray pump.

Checks of the reactor head vent and relief valve position indicating lights each shift confirm control power available for actuation.

LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the instrumentation and associated devices which initiate a reactor scram.

Objective

To assure the operability of the reactor protection system whenever irradiated fuel is in the reactor vessel.

Specification

A. Whenever irradiated fuel is in the reactor vessel the setpoints and minimum number of channels required to be functional are listed in Table 3.1.A/4.1.A.

SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification

A. Instrument channels shall be functionally tested and calibrated as indicated in Table 3.1.A/4.1.A.

TABLE 3.1.A/4.1.A

REACTOR PROTECTION SYSTEM INSTRUMENTATION

Minimum Number of Functional Channels per Trip System	Trip Function	Trip Level Setting	Functional Test Frequency	Alarm Level Setting	Calibration Frequency
1	Manual scram	N/A	once/month	N/A	N/A
2 (1) (2)	SRM high-high count	$\leq 5 \times 10^5$ cps	once/week	$\leq 5 \times 10^5$ cps (upscale HI HI) $\leq 10^5$ cps (upscale HI) ≥ 3 cps (downscale)	once/3 mos
2 (3)	High Water Level in Scram Discharge Tank	N/A	once/3 months	$\leq$ 50 Gallons	once/3 month

Notes for Table 3.1.A/4.1.A

(1) The SRM's are presently connected in the non-coincident mode in the RPS, where any 1 SRM upscale HI HI will initiate a scram. Of the 4 SRM channels, only 2 are required to be functional.

(2) It is recognized that SRM count rate will diminish and eventually disappear during the process of removing fuel from a reactor at which time there is no required proof of functionability.

(3) Channels A or C and B or D required.

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3.1/4.1 BASES: REACTOR PROTECTION SYSTEM

The reactor protection system will be available for rod insertion in the unlikely event that a rod becomes withdrawn. This is highly improbable because the reactor mode switch is locked in the shutdown position and the control rod directional solenoids are electrically disarmed. The reactor protection system is functional as described in Section 7.2 of the FSAR with the manual scram and the SRM's required functional. The SRM's have been placed in the non-coincidence scram mode to provide automatic scram from high-high count rate.

The two level switches and their associated alarms provide assurance that an adequate volume exists in the scram discharge volume tank to accommodate a scram.

LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the plant instrumentation which initiates and controls a protective function.

Objective

To assure the operability of protective instrumentation.

Specification

A. Surveillance Instrumentation

The limiting conditions for the instrumentation that provides surveillance information required in the period during which these specifications are effective are given in Table 3.2.A.

1. Reactor water level shall be maintained at ≥ 555" above vessel zero whenever irradiated fuel is in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENTATION

Applicability

Applies to the surveillance requirement of the instrumentation that initiates and controls protective function.

Objective

To specify the type and frequency of surveillance to be applied to protective instrumentation.

Specification

A. Surveillance Instrumentation

The surveillance requirements associated with the equipment listed in Table 3.2.A shall be that indicated in Table 4.2.A.

1. The reactor water level shall be verified once each shift.

Table 3.2.A

SURVEILLANCE INSTRUMENTATION

Minimum # of Operable Instrument Channels	Instrument #	Instrument	Type Indication and Range	Alarm Setting	Notes
2	LI-3-206 or LR-3-53 or LI-3-53 or LI-3-55 and LI-3-46A or 46B	Reactor Water Level	Indicator 0" to 60" Recorder 0" to 60" Indicator 0" to 60" Indicator 0" to 400" Indicator +60" to -155"	Low ≥ 27", high ≤39"	(1)(4)
2	PI-3-54 PR-3-53	Reactor Pressure	Indicator 0-1200 psig Recorder 0-1200 psig	High ≤ 1040 psig	(1)(5)
2	PR-64-50 and PI-64-67	Drywell Pressure	Recorder 0-80 psig Indicator 0-80 psig		(1)(5)
2	TI-64-52A and TR-64-52	Drywell Temperature	Indicator 0-400° F. Recorder 0-400° F.	High ≤ 145° F.	(1)(5)
2	TI-64-55A and TIS-64-55	Suppression Chamber Water Temperature	Indicators 0-400° F.	High ≤ 90° F	(1)(4)
1	LI-64-54A or LI-64-66	Suppression Chamber Water Level	Indicator -25" to +25"		(1)(4)
1	NA	Control Rod Position	Continuity		(2)(4)
2	SRM A, B, C, D	Neutron Monitoring	Indicator and Recorder 0.1 to 10^{+6} cps -100 to +10 sec. (period)	Downscale ≥ 3 cps Retract permit ≥ 100 cps Upscale HI ≤ 10^5 cps Upscale HI-HI ≤ 5×10^5 cps Period ≥ 30 sec.	(1)(3)(4)
1	LS-78-2A	Fuel Storage Pool level high	NA	EL 663' 1 1/2"	(6)
1	LS-78-2B	Fuel Storage Pool level low	NA	EL 662' 7 1/2"	(6)
1	TR-74-80 pT 17	Fuel pool temperature	Recorder 0-600°F	≤ 125°F	(6) (7)

NOTES FOR TABLE 3.2.A

(1) If one of the instrument channels monitoring a parameter should become incapable of performing its intended function, all operations which could affect the associated system will be suspended until the item is returned to service.

(2) The control rod position indicator full-in switches will be operable for every control rod and will provide indication in the control room, or the control rod position will be verified to be full-in by a continuity check of the full-in limit switches at the local panel. Prior to the unbolting of the reactor vessel head, control rod position indication to verify full-in position will be established in the control room and maintained thereafter when fuel is in the vessel.

(3) The following Source Range Monitoring Channels will be operable and will provide count rate indication and alarms in the control room:

SRM Channel A or C

SRM Channel B or D

The alarms in the control room will be as follows:

SRM TRIPS

Trip Function	Trip Action
SRM upscale or inoperative	Annunciator, amber light
SRM Downscale	Annunciator, white light

In addition, the SRM's have been placed in the non-coincidence scram mode to provide for protection against high neutron flux. It is recognized that SRM count rate will diminish and eventually disappear during the process of removing fuel from the reactor at which time there is no required proof of functionability.

(4) Only required whenever irradiated fuel is in the reactor vessel.

(5) Only required whenever the reactor pressure vessel head is bolted in place.

(6) Only required when irradiated fuel is in the fuel pool.

(7) If this instrument channel is out of service, alternate manual means will be used until it is returned to service.

MINIMUM TEST AND CALIBRATION FREQUENCY FOR SURVEILLANCE INSTRUMENTATION (POST ACCIDENT INSTRUMENTATION)

	Instrument Channel	Calibration Frequency (3)	Instrument Check
1)	Reactor Water Level	Once/6 months	Each Shift
2)	Reactor Pressure	Once/6 months	Each Shift
3)	Drywell Pressure	Once/6 months	Each Shift
4)	Drywell Temperature	Once/6 months	Each Shift
5)	Suppression Chamber Water Temperature	Once/6 months	Each Shift
6)	Suppression Chamber Water Level	Once/6 months	Once/Week (4)
7)	Control Rod Position	NA	Each Shift (1)
8)	Neutron Monitoring (SRM)	Once/3 months (2)	Each Shift
9)	Drywell Pressure (PS-64-67)	Once/6 months	NA
10)	Fuel Pool Level (LS-78-2A, 2B)	Once/6 months	Once/Day
11)	Fuel Pool Temperature (TR-74-80)	Once/6 months	Once/Day

NOTES FOR TABLE 4.2.A

(1) Control rod position indication to verify the full-in position will be maintained in the control room on Unit 2. While the reactor vessel head is bolted in place unit 1 control rods which do not have position indication in the control room will be monitored at the local panel by use of the full-in limit switches at intervals not to exceed once/3 days. Prior to and subsequent to the unbolting of the reactor vessel head, control rod position indication to verify the full-in position will be maintained in the control room. This monitoring circuitry will be tested once a week.

(2) The SRM will be functionally tested on frequency of once per week.

(3) Instrument calibration will be verified upon restoration, then at the frequency listed.

(4) Instrument check will be to observe instrument response as transmitter is removed and returned to service.

3.2/4.2 BASES: PROTECTIVE INSTRUMENTATION

For each parameter monitored, as listed in Table 3.2.A, there are equivalent or redundant channels of instrumentation as noted. By comparing readings between two channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will necessitate an early recalibration, thereby maintaining the quality of the instrument readings.

The redundant monitoring of reactor vessel level will be sufficient to provide the operator with information with which to determine the adequacy of the coolant inventory. The approach of abnormal conditions will be brought to the operator's attention by audible alarms.

Instrumentation has been selected to provide control room indication of adequate information regarding status of the drywell pressure and temperature and torus water level and temperature. Monitoring of information concerning key primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

A minimum of two channels of the SRM's, each on separate power supplies, will be operable. Each channel will provide indication of count rate and provide alarms as listed in Table 7.5-1 of the FSAR.

An automatic scram from high-high count rate of any SRM's channel as well as a manual scram will be available. One control rod drive pump each for units 1 and 2 will be kept in operation and all the accumulators will be kept charged with nitrogen and water. The scram pilot valves and pneumatically controlled valves to scram each control unit will be operable.

3.2/4.2 BASES: PROTECTION INSTRUMENTATION (CONTINUED)

Control rod position indication to verify the full-in position will be maintained on unit 2. The continuity of all the full-in limit switches on unit 1 will be monitored periodically. Under the conditions to be maintained in Temporary Technical Specification 3.3, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions. Prior to removing the reactor vessel head bolts, a monitoring system will be provided in the main control room to alert the operator if any rod drift should occur.

The reactor water level will be nominally maintained at a level greater than 27" but less than 39" as indicated on LR-3-53. Above or below this level will give an alarm of "Reactor Water Level Abnormal." 27" indication on LR-3-53 corresponds to 555" above vessel zero.

In the unlikely event that loss of fuel pool level occurs, the low level alarm is set high enough above the fuel such that operator action can be taken to establish makeup before the level reaches 8-1/2 feet above the fuel. Refer to the specification 3.10 for fuel pool water conditions.

LIMITING CONDITIONS FOR OPERATION

3.3 REACTIVITY CONTROL

Applicability

Applies to the operational status of the control rod system.

Objective

To assure the ability of the control rod system to control reactivity while fuel is in the reactor vessel.

Specification

While fuel is in the reactor vessel the requirements of 3.3.A through 3.3.G shall be met.

A. All control rods shall be inserted in the full-in position.

B. The directional control valves shall be disarmed electrically for all control rods.

C. The manual valves in the drive water supply shall be in the shut position to prohibit rod movement.

D. The control rod accumulators shall be charged.

E. Two SRM channels shall be functional.

F. One control rod drive pump shall be in service.

SURVEILLANCE REQUIREMENTS

4.3 REACTIVITY CONTROL

Applicability

Applies to the surveillance requirements of the control rod system.

Objective

To verify the ability of the control rod system to control reactivity.

Specification

A. Control rod position shall be verified in accordance with Table 4.2.A.

B. Each directional control valve shall be verified to be electrically disarmed at intervals not to exceed once every 3 days.

C. The drive water supply valve (85-593) to each hydraulic control unit shall be verified closed and the water supply valves (85-612, 85-613) to each shall be verified open at intervals not to exceed once every 3 days.

D. The accumulator pressure shall be checked once a day.

E. The count rate shall be recorded once each shift.

F. The control rod drive pump discharge pressure shall be checked once per shift.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.3 REACTIVITY CONTROL Specification G. Prior to unbolting the reactor vessel head, and thereafter while fuel is in the reactor when normal control rod position indication is not available, the continuity of the control rod full-in limit switches shall be monitored continuously with visual indication and audible alarm in the control room to indicate that any control rod is not fully inserted or that the monitoring circuit is interrupted or deenergized.	4.3 REACTIVITY CONTROL Specification G. The temporary circuit provided to monitor control rods without position indication shall be functionally tested prior to unbolting the reactor vessel head and at intervals not to exceed once per week.

To prevent an inadvertent or spurious withdrawal of a control rod the directional control valves of each control rod have been electrically disarmed. As a further precaution, the valve in the drive water supply to each hydraulic control unit will be closed. In the unlikely event that a control rod does become withdrawn, two channels of the SRM's are required to be available for visual indication of neutron level. Although the SRM's may not immediately respond to a single rod movement, they would be adequate to monitor the approach to criticality. Also the SRM's are connected in the non-coincidence scram mode to provide rapid rod insertion from a high-high count rate on either SRM. Additionally, a manual scram will also be available. To secure that the control rods can be scrammed, the control rod accumulators are required to be charged with nitrogen and water pressure and a control rod drive pump is required to be in service. To provide additional indication of a control rod withdrawal, the control rod position indicator full-in switches will be functional for every control rod and will provide indication in the control room except, when the reactor vessel head is bolted in place, the control rod position may be verified to be full-in by a continuity check of the full-in limit switch at the local panel. Prior to unbolting the reactor vessel head, circuitry will be provided to permit continuous monitoring in the control room of continuity through the full-in limit switches when normal indication is not available. The circuitry will be testable and will provide a light and audible alarm in the control room to assure operator attention should any one rod be moved from the fully inserted position. Once the fuel is removed from the reactor vessel, the control of reactivity is assured by the design of the spent fuel storage pool as described in FSAR, Chapter 10, Subsection 3.

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LIMITING CONDITIONS FOR OPERATION

4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Standby Liquid Control System.

Objective

To assure the availability of a system with the capability to shut down the reactor and maintain the shutdown condition without the use of control rods.

Specification

A. Normal System Availability

1. The standby liquid control system shall be functional whenever fuel is in the reactor vessel.

2. The injection line manual isolation valve 63-524 will will be closed except when the system is required for emergency injection.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the surveillance requirements of the Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal System Availability

The operability of the Standby Liquid Control System shall be verified by the performance of the following tests:

1. At least once per month each pump loop shall be functionally tested.

2. At least once during each operating cycle:

   a. Check that the setting of the system relief valves is 1425 ± 75 psig.

   b. Manually initiate the system, except explosive valves. Pump boron solution through the recirculation path and back to the Standby Liquid Control Solution Tank. Minimum pump flow rate of 39 gpm

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
.4 STANDBY LIQUID CONTROL SYSTEM	4.4 STANDBY LIQUID CONTROL SYSTEM
	against a system head of 1275 psig shall be verified. After pumping boron solution, the system shall be flushed with demineralized water.
	c. Manually initiate one of the Standby Liquid Control System loops and pump demineralized water into the reactor vessel.
	This test checks explosion of the charge associated with the tested loop, proper operation of the valves, and pump operability. Replacement charges shall be selected such that the age of charges in service shall not exceed five years from the manufacturers assembly date.
	d. Both systems, including both explosive valves, shall be tested in the course of two operating cycles.
B. Operation with Inoperable Components:	B. Surveillance with Inoperable Components:
1. From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled if the inoperable component is returned to service within seven days.	1. When a component is found to be inoperable, its redundant component shall be demonstrated to be operable immediately and daily thereafter until the inoperable component is repaired.

LIMITING CONDITIONS FOR OPERATION

.4 STANDBY LIQUID CONTROL SYSTEM

C. Sodium Pentaborate Solution

At all times when the Standby Liquid Control System is required to be operable the following conditions shall be met:

1. The net volume - concentration of the Liquid Control Solution in the liquid control tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2. This includes the piping between the standby liquid control tank and the suction inlet to the pumps.

SURVEILLANCE REQUIREMENTS

4.4 STANDBY LIQUID CONTROL SYSTEM

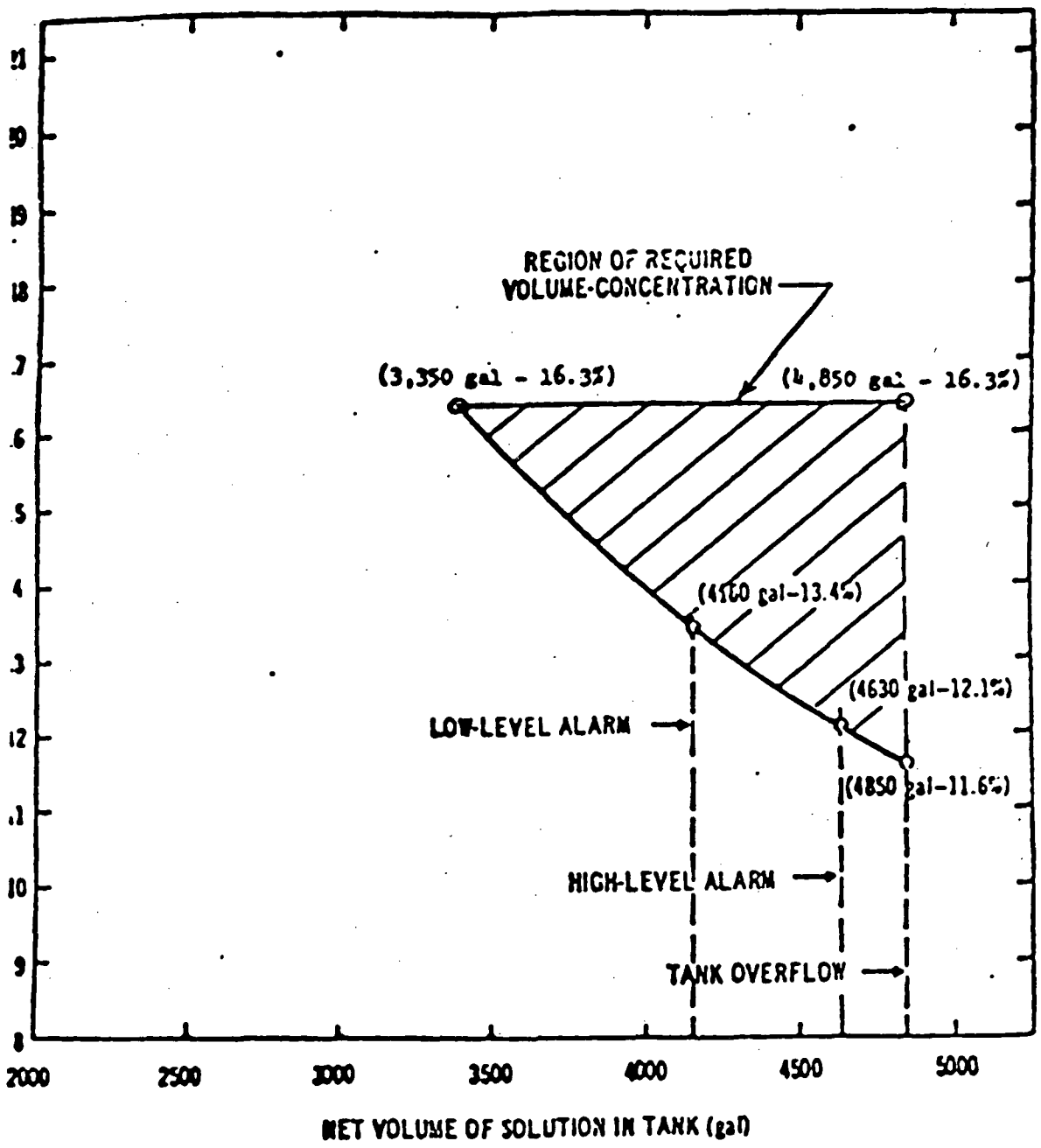
C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check at least once per day.

2. Temperature: Check at least once per day.

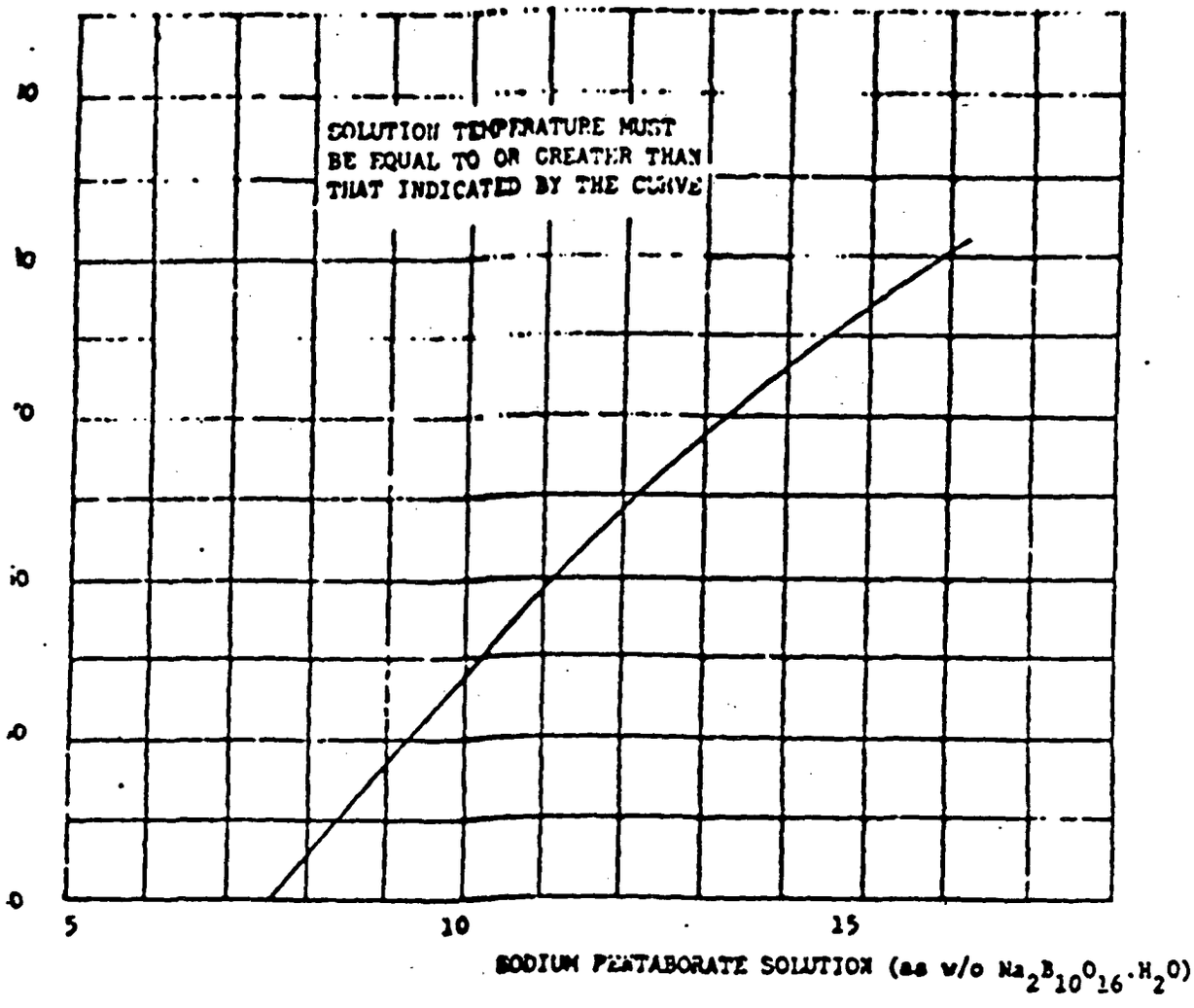
3. Concentration: Check at least once per month. Also check concentration any time water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
VOLUME-CONCENTRATED REQUIREMENTS

FIGURE 3.4-1 JUN 1 3 1975



BROWNS FERRY NUCLEAR PLANT
FINAL SAFETY ANALYSIS REPORT

SODIUM PENTABORATE SOLUTION
TEMPERATURE REQUIREMENTS

FIGURE 3.4-2

BASES: STANDBY LIQUID CONTROL SYSTEM

A. The purpose of the liquid control system is to provide the capability of maintaining the reactor in a cold shutdown condition assuming that none of the withdrawn control rods can be inserted. To meet this objective, the liquid control system is designed to inject a quantity of boron that produces a concentration greater than 600 ppm of boron in the reactor core in less than 125 minutes.

The minimum limitation on the relief valve setting is intended to prevent the loss of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve settings provides system protection from overpressure.

B. Only one of the two standby liquid control pumping loops is needed for operating the system. Assurance that the remaining system will perform its intended function and that the long-term average availability of the system is not reduced is obtained from a one-out-of-two system.

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a possible solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10° F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

BASES: STANDBY LIQUID CONTROL SYSTEM (Continued)

C. (Continued)

The volume concentration requirements of the solution are such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature-concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 4,160 gallons of solution having a 13.4 percent sodium pentaborate concentration or the equivalent is required to meet this shutdown requirement as defined in Figure 3.4.1.

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

LIMITING CONDITIONS FOR OPERATION

.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the operational status of the core, suppression pool, and fuel pool cooling systems.

Objective

To assure the operability of the core and containment cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. When irradiated fuel is in the reactor vessel, at least one core spray loop with one pump and available diesel generator shall be capable of delivering flow.

B. Residual Heat Removal System (RHRS) Containment and Shutdown Cooling

1. When irradiated fuel is in the reactor vessel at least two RHR loops with one pump per loop

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

Applicability

Applies to the surveillance requirements of the core, suppression pool, and fuel pool cooling systems when the corresponding limiting conditions for operation are in effect.

Objective

To verify the operability of the core and suppression pool cooling systems under all conditions for which this cooling capability is an essential response to plant abnormalities.

Specification

A. Core Spray System (CSS)

1. Core Spray System Testing

Item	Frequency
a. Pump operability	upon restoration and monthly thereafter
b. Motor operated valve operability	upon restoration and monthly thereafter

2. When it is determined that one core spray loop with one pump and available diesel generator is incapable of deliverin

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

shall be capable of manual operation. Diesel generators must also be available to power the pumps. A service water supply must be available.

C. Spent Fuel Pool Cooling System

1. Whenever irradiated fuel is stored in the spent fuel pool, the cooling system for the spent fuel pool shall be capable of maintaining the temperature of the fuel pool coolant ≤125°F.
2. When irradiated fuel is stored in the spent fuel pool, except as specified in 3.5.C.3 two fuel pool cooling pumps and heat exchangers shall be capable of maintaining fuel pool temperatures as specified in 3.5.C.1. Diesel Generators must also be available to power the pumps. The associated RBCCWS loop and service water system must also be operating.

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

flow, another core spray pump with an available diesel generator shall be selected, and all active components in the flow paths shall be immediately demonstrated to be capable of delivering flow.

B. Residual Heat Removal System (RHR) (Containment and Shutdown Cooling)

1. Residual Heat Removal System Testing

	Item	Frequency
a.	Pump operability	Upon restoration and monthly thereafter
b.	Motor operated valve operability	Upon restoration and monthly thereafter

2. When it is determined that one RHR pump (containment and suppression pool cooling) or associated heat exchanger is incapable of delivering flow and removing heat at a time when flow capability and heat removal are required, the remaining RHR pump and associated heat exchanger and available diesel generator, and all active components in the flow paths

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3. Whenever the temperature of the water in the spent fuel pool exceeds 125°F or one of the fuel pool cooling systems is made or becomes unavailable, the RHR supplemental fuel pool cooling mode shall be operable or operating if necessary to maintain the fuel pool temperature less than 125°F.	
4. Whenever irradiated fuel is stored in the fuel pool, the gates on the fuel transfer canal between Unit 1 and Unit 2 shall be left in place and the transfer canal drain valves 1-78-561 and 1-78-562 shall be shut; except if a Unit fuel pool cleanup system becomes unavailable, the transfer canal between Unit 1 and Unit 2 may be opened allowing the other Unit fuel pool cleanup system to be used for both units.	

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

D. RHR Spent Fuel Pool Supplemental Cooling

1. Whenever irradiated fuel is stored in the spent fuel pool at least one RHR loop with one RHR pump capable of being supplied with diesel power shall be capable of manual operation in the fuel pool supplemental cooling mode. The associated RHRSW pump, heat exchanger must also be functional.

E. RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

1. When irradiated fuel is in the reactor vessel, 4 RHRSW pumps must be functional and aligned to RHR header service corresponding to the 4 selected RHR pumps.

2. At all times, at least 2 RHRSW pumps shall be assigned to

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

capable of delivering flow and heat removal immediately and weekly thereafter until the inoperable RHR pump and associated heat exchanger is returned to service or an alternate pump and heat exchanger with an available diesel generator selected and verified.

C. Spent Fuel Pool Cooling System

1. The spent fuel pool water temperature shall be checked and recorded at least every 24 hours.

2. Prior to opening the transfer canal between the Unit 1 and Unit 2 fuel pools, and daily thereafter, verify that drain valves 1-78-561 and 1-78-562 are shut.

D. RHR Spent Fuel Pool Supplemental Cooling

1. Routine surveillance for the RHR pump is specified 4.5.B.1.

2. When it is determined that the selected RHR pump for spent fuel pool supplemental cooling is incapable of heat removal, another RHR pump capable of

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT AND FUEL POOL COOLING SYSTEMS

EECW header service with one pump assigned to each header. Each pump must run continuously with its loss of voltage trip deactivated, or it shall be capable of automatic start in its normal D/G load sequencing mode of operation. Each pump shall be assigned to a separate diesel power supply.

3. Whenever irradiated fuel is stored in the reactor vessel one RHRSW pump (D1 or D2) capable of being supplied with diesel power normally or alternately assigned to the RHR heat exchanger header supplying the standby coolant supply connection must be available with a flow path to the reactor vessel.

4. Whenever irradiated fuel is stored in the spent fuel pool, two independent flow paths for water make up to the spent fuel pool will be available from two RHRSW pumps, capable of being supplied by diesel power.

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

being supplied with diesel power shall be selected, and all active components required for heat removal capability, shall be demonstrated to be capable of delivering flow.

3. The RHR pump and valves used for fuel pool supplemental cooling shall be tested upon restoration and monthly thereafter.

4. If requirements 4.5.D.1 and 4.5.D.2 cannot be met, the requirements of specification 4.5.E.5 shall be met.

E. RHR Service Water System (RHRSWS) And Emergency Equipment Cooling Water System (EECWS)

1. RHR Service Water System

Each of the four RHRSW pumps and associated essential control valves on the RHR heat exchanger headers shall be demonstrated to be functional upon restoration and once every three months thereafter if not in continuous service.

LIMITING CONDITIONS FOR OPERATION

3.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

F. Maintenance of Filled Discharge Pipe

Whenever the core spray system or RHR systems are required to be functional, the discharge piping from the pump discharge of these systems to the last block valve shall be filled. The condensate head tank shall be aligned to serve the discharge piping of the RHR and CS pumps. The pressure indicators on the discharge piping of the RHR and CS pumps shall indicate not less than listed below.

PI-75-20	70 psig
PI-75-48	70 psig
PI-74-51	70 psig
PI-74-65	70 psig

SURVEILLANCE REQUIREMENTS

4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

2. EECW System

Each of the 2 RHRSW pumps assigned to EECW service will be in continuous service. Associated essential control valves on the EECW headers shall be demonstrated to function once every three months.

3. When it is determined that one RHRSW pump and associated control valves on an RHR heat exchanger header is incapable of delivering flow at a time when flow delivery capability is required, the remaining RHRSW pumps and associated heat exchangers and available diesel generators, and all active components in the flow paths shall be demonstrated to be capable of delivering flow immediately and weekly thereafter until the inoperable RHRSW pump is returned to service or an alternate pump is selected and verified.

4. When it is determined that one RHRSW pump and associated

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS
	control valves on an EECW header is incapable of delivering flow at a time when flow delivery capability is required, an alternate THRSW pump on a corresponding diesel generator shall be selected and assigned to the same EECW header.
	5. At intervals not to exceed 7 days each independent fuel pool makeup flow path from its respective EECW header to the fire hose outlet connection on the refueling floor will be tested to verify makeup water supply availability.
	F. Maintenance of Filled Discharge Pipe The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray system and RHR system are filled:

3.10 CORE ALTERATIONS

H. Restoration Work

1. Removal of fire damaged equipment shall not be initiated until all fuel from both cores is stored in their fuel storage pools and the fuel pools and the fuel pool gates between the fuel pools and the reactor cavities are installed with the canal blocks in place.

2. After all fuel from both cores is stored in its respective fuel storage pool and the fuel pool gates are installed with canal blocks in place, removal of fire damaged equipment may be carried out in accordance with the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2" and revisions thereto up to and including Revision 10 as supplemented by licensee's letter dated June 11, 1975 (Gilleland to Rusche, NRC)

3.10/4.10 BASES: CORE ALTERATIONS

A. Refueling Reactivity Controls

The electrical circuits to the insert and withdraw directional control valves of each control rod drive hydraulic control unit will be disconnected at the valves to prevent inadvertent or spurious withdrawal of a control rod during fuel removal. As a further precaution, the manual valve in the drive water supply to each hydraulic control unit will be closed.

Each control rod is mechanically latched in the full-in position and withdrawal would require a prescribed set of operations to (1) relieve pressure on the latch by a slight insertion, (2) release the latch, and (3) withdraw the rod. The likelihood of the required operations occurring asthe result of any spurious, inadvertent, or accident condition is so small as to be incredible. The control rod drives are seismically qualified. Breaking or cutting any hydraulic line will not cause a control rod withdrawal. The full-in positions of all the control rods in both units will be indicated, or monitored, continuously.

Under the conditions to be maintained, the probability of a control rod withdrawal is significantly lower than that following a scram from normal conditions.

Redundant SRM alarms in the control room would alert the operator of significant reactivity increase, and automatically initiate a scram. Manual scram capability is maintained while fuel is in the reactor.

The refueling interlocks are designed to back up procedural core reactivity controls during refueling operations. By sensing

A. Refueling Reactivity Controls (Continued)

the condition of the refueling equipment and the control rods, the interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlock. These interlocks are activated by placing the mode switch in the refueling position. When the mode switch is in the refuel position, one control rod can be withdrawn.

By requiring that the mode switch remain in the shutdown position while removing fuel from the reactor vessel, control rod withdrawal is prohibited. In addition, the insert and withdrawal directional control valves of each control rod drive hydraulic control unit will be disconnected at the valves to prevent inadvertent or spurious withdrawal of a control rod, and the valve in the drive water supply to each hydraulic control unit will be closed. Since the ability to withdraw a control rod is not present, the need for the refueling interlocks no longer exists.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.5 CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEM 1. Prior to the operation of the RHRS and core spray systems, the discharge piping of these systems shall be vented from the high point and water flow determined. 2. Following any period where the RHR or core spray systems have not been required to be operable, the discharge piping of the nonfunctional system shall be vented from the high point prior to the return of the system to service. 3. When the RHRS and CSS are required to be functional the pressure indicators which monitor the discharge lines shall be monitored daily and the pressure logged.

3.5.A Core Spray System (CSS)

The core spray loop provides one of several means of supplying water to the reactor vessel. The capacity of one core spray pump is in excess of 10 times the amount of water required to cool the reactor core in its present shutdown condition. Power operation of the reactor will not be permitted, therefore, the margin available for this purpose will increase with time. Technical Specifications 3.5.A and 3.5.B require that at least three pumps will be available to assure that adequate cooling of the fuel is accomplished. However, considerably more equipment is available, including additional core spray, RHR, control rod drive, condensate booster, hotwell and raw water pumps.

Analyses have been performed to demonstrate that in excess of 25 hours is available to restore core cooling if for some unforeseen reason cooling is interrupted. Studies were performed to show that a core spray loop could be manually placed in operation in less than one hour. Hence, considerable margin is available for manual actuation.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.B Residual Heat Removal System (RHRS) (Containment and Shutdown Cooling)

The decay heat removal requirements for one unit in the cold shutdown condition can be conservatively met by the operation of one RHR pump and its associated RHR heat exchanger in the shutdown cooling mode. The total heat load for the heat exchanger is estimated to be less than one-fourth of the heat exchanger capability under the required flow and temperature conditions. At least three RHR pumps and associated heat exchangers on each unit are capable of delivering flow and removing the decay heat. The low decay heat and absence of pressure which could foster an unacceptable loss of coolant allows ample time for manual operation in accordance with established operating instructions.

3.5 NAMES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.C Spent Fuel Pool Cooling System

The spent fuel pool cooling system consists of two [illegible] type pumps and heat exchangers. Figure D-2 of the TVA Safety Analysis of the BFNP Units 1 and 2 included as Part VI, Section E, of the "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Five)" shows that the system is capable of maintaining the spent fuel coolant temperature below 125°F based on the ANS standard 3-year operation decay heat curve. One pump and heat exchanger will provide sufficient cooling capability to prevent pool boiling.

In response to NRC Question 10.1, dated March 25, 1971, TVA committed to modify and upgrade the spent fuel pool cooling system to qualify as a Seismic Class I system. The system was subsequently analyzed and designed in accordance with the requirements of ANSI B 31.1.0, 1967. The loading combinations and allowable stresses used in the analysis were in accordance with ASME Section III, Subsection NC, 1971 requirements. The analysis meets the intent and requirements of ASME Section III, 1974. All piping, valves, and equipment as shown in Figure Q10.1-1 of the response, except that identified as nonseismic, were analyzed.

The gates on the fuel transfer canal between Unit 1 and Unit 2 will be left in place and the transfer canal drain line will be valved out. This provides redundant seismically qualified barriers for the prevention of pool leakage through the transfer canal. The transfer canal connecting the two pools may be opened allowing each fuel pool clean-up system to provide redundancy for the other in the event of a Unit Fuel Pool Clean-up System failure provided the canal drains are verified closed.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.D RHR Spent Fuel Pool Supplemental Cooling

The decay heat removal requirements for a full core stored in the fuel pool can be conservatively met by the operations of one RHR pump and its associated RHR heat exchanger in the fuel pool cooling mode. The total heat load for this mode is estimated to be less than 60 per cent of the heat exchanger capability under the required flow and temperature conditions.

During the original preoperational test program, each RHR pump was operated in the fuel pool cooling mode of operation. The test included operation of the RHR system (1000-2000 gpm) in parallel with the fuel pool cooling system. Two fuel pool cooling pumps were operated with 600 gpm per pump. This test verified the ability to operate the RHR system in parallel with the fuel pool cooling system at design flow rates.

One RHR pump and associated RHRSW pump capable of being supplied with diesel power will be functional any time spent fuel is stored in the spent fuel pool.

3.5 BASES: CORE, CONTAINMENT, AND [illegible] COOLING SYSTEMS

3.5.2 RHR Service Water System (RHRSWS) Emergency Equipment Cooling Water System (EECWS)

The decay heat removal cooling water requirements for two units in the cold shutdown condition can be conservatively met by the operation of one RHRSW pump on one heat exchanger on each unit. One RHRSW pump is required for each unit if the units are using heat exchangers which are not on the same service water header. Four RHRSW pumps are presently available and capable of delivering flow to meet this requirement. Less than one-half the flow delivery capability of each pump is needed to remove the present decay heat for each unit. The low decay heat level and ample flow delivery capability allow ample time for manual operation in accordance with established operating instructions.

The standby emergency equipment cooling water (EECW) requirements for two units in the cold shutdown condition can be adequately met by the operation of one RHRSW pump on an EECW header. The EECW system is not required for normal plant shutdown operation because the required cooling water is supplied by the plant raw cooling water system. The principal immediate need for EECW flow is in the event that a diesel engine should be started. In this case, EECW flow must be established at once. To meet this requirement, two RHRSW pumps are assigned to EECW service are running continuously with each aligned to a separate supply header. Each of these pumps are operating continuously (with loss of voltage trips deactivated) or they must be capable of automatic start in their normal diesel generator load sequencing mode of operation. Either header will supply all EECW requirements. In addition, two other RHRSW pumps are available and capable of delivering flow by manual operations in accordance with established operating instructions. The two RHRSW pumps which are running continuously or have automatic restarting capability are assigned to 4.16 kV shutdown boards which have associated operable diesel generators and the pumps are

3.5.E RHR Service Water System (RHRSW) Emergency Equipment Cooling Water System (EECWS) (Continued)

arranged with automatic restart provisions in the event that board undervoltage should occur.

In the unlikely event that all make up capability is lost, water can be supplied to the reactor or fuel pool by certain RHRSW pumps directly from the river. The D1 or D2 RHRSW pumps can pump through the RHRSW header and standby coolant supply line in to RHR loop II on Unit 1, or RHR loop I on Unit 2. From the RHR loop the water can be routed to the reactor through the LPCI injection valves or the fuel pool through the fuel pool system connections on each RHR loop. An alternate path is available which is independent of the RHR and fuel pool cooling systems. The alternate path will only require any one of four RHRSW pumps and manual valve operation to provide make up coolant directly to the fuel pool through a hose connected to either the north or south EECW header hydrant. The RHRSW pump has an on site power source. The make up capability provided by the RHRSW pump far exceeds the amount needed to replace the water lost at the maximum evaporation rate possible with the present decay heat.

3.5 BASES: CORE, CONTAINMENT, AND FUEL POOL COOLING SYSTEMS

3.5.F Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray and RHR system are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Tehcnical Specification requires the discharge lines to be filled whenever the system is in a functional condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be nonfunctional for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow prior to any pump operation to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a head tank located approximately 100 feet above the discharge line high point supplies makeup water for these systems. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and approximately 70 psig for a water level at the head tank and are monitored daily to ensure that the discharge lines are filled.

3.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the status of the reactor coolant system.

Objective

To assure the integrity of the reactor coolant system.

Specification

A. Thermal and Pressurization Limitations

The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100° F.

B. Coolant Chemistry

Reactor water shall be maintained within the following limits: Conductivity ≤10 umhos/cm @ 25° C. and Chloride ≤0.5ppm.

C. Safety and Relief Valves

Four safety relief valves which have accumulators shall be functional on each unit with two valves powered from each of two 250V DC MOV boards when the reactor vessel head is bolted in place.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification

A. Thermal and Pressurization Limitation

When the reactor vessel head bolting studs are tensioned and the reactor is in a cold condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

B. Coolant Chemistry

1. A sample of reactor coolant shall be analyzed:
   a. at least every 96 hours for conductivity and chloride ion content.
   b. at least every 8 hours for conductivity and chloride ion content when the continuous conductivity monitor is inoperable.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6 PRIMARY SYSTEM BOUNDARY	4.6 PRIMARY SYSTEM BOUNDARY
D. Reactor Vessel Head Vent The reactor vessel head vents shall be in the open position when the reactor vessel head is bolted in place and the head vent piping is assembled.	C. Safety and Relief Valves The position indicating lights for the four safety relief valves specified in 3.6.C.1 shall be checked each operating shift to confirm control power available for actuation when the reactor vessel head is bolted in place. D. Reactor Vessel Head Vent The position indicating lights on the reactor vessel head vents shall be checked each operating shift to confirm the valves are indicated to be in the open position when the reactor vessel head is bolted in place and vent piping is assembled.

3.6.A/4.6.A BASES: THERMAL AND PRESSURIZATION LIMITATIONS

Tightening the studs on the reactor vessel head flexes it slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The reactor vessel head flange and head are constructed such that their initial maximum NDTT is 10° F, as cited in paragraph 4.2.7 of the safety analysis report. Therefore, the initial minimum temperature at which the studs can be placed in tension is established at 40° F + 60° F or 100° F.

3.6.B/4.6.B BASES: COOLANT CHEMISTRY

Materials in the primary system are primarily 304 stainless steel and the Zircaloy cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is continuously measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations and dissolved oxygen. Zircaloy does not exhibit similar stress corrosion failures.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. Chloride measurements are made to determine whether or not they are also out of their normal operating values because conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride. In some types of water-cooled reactors, conductivities are in fact high due to purposeful addition of additives. In the case of BWR's, however, where no additives

3.6.B/4.6.B BASES: COOLANT CHEMISTRY (Continued)

are used and where near neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded.

Methods available for correcting the off-standard conditions include using the cleanup demineralizer, reducing the input rate of impurities, or changing the supply of cooling water. The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hours will serve as a reference for calibration of these monitors and are considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Reactor coolant sampling is increased to once per shift when the continuous conductivity monitor is unavailable.

.6.C/4.6.C BASES: SAFETY AND RELIEF VALVES

With the reactor in the cold shutdown condition and with low decay heat, four safety relief valves are more than adequate to assure the integrity of the reactor coolant systems. Control power from different 250V MOV boards assures availability of two valves even in the unlikely event of loss of a 250V MOV board. Surveillance checks each shift assure control power is available for valve actuation. When the reactor vessel head is not bolted in place, the relief valves are not required.

4.6.D/4.6.D · [illegible] REACTOR VESSEL HEAD VENT

Maintaining the reactor vessel head vents in the open position with the RHR shutdown cooling system in service keeps the reactor vessel at atmospheric pressure when in the cold shutdown condition. Surveillance checks of the position indicating lights are made each shift.

When the reactor vessel head is not bolted in place or the vessel head is removed or the head vent piping is removed from the head, adequate venting capability exists.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability

Applies to the status of the primary containment systems and refuel zone ventilation.

Objective

To monitor pressure and temperature of the primary containment system and provide refuel zone ventilation.

Specification

A. [illegible]

1. [illegible]

2. [illegible] well [illegible] ture shall be monitored. (See Table [illegible])

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability

Applies to the primary containment and refuel zone ventilation.

Objective

To verify the pressure and temperature of the primary containment and assure air exhaust from the refuel zone.

Specification

A. Primary containment

1. The suppression chamber water level and temperature shall be checked once per 24-hours when fuel is in the reactor.

2. The drywell pressure and temperature shall be checked once per 24-hours when the reactor vessel head is bolted in place.

LIMITING CONDITIONS FOR OPERATIONS	SURVEILLANCE REQUIREMENTS
3.7 CONTAINMENT SYSTEMS	4.7 CONTAINMENT SYSTEMS
	Specification
B. REFUEL ZONE VENTILATION	B. REFUEL ZONE VENTILATION
When the reactor vessel head is unbolted, and when the fuel is stored in the fuel pool, there shall be one refueling floor exhaust fan and one supply fan in service per unit.	When the reactor vessel head is unbolted operation of one refueling floor supply fan and one exhaust fan shall be verified and documented once per shift.
C. STANDBY GAS TREATMENT	C. STANDBY GAS TREATMENT
When the reactor vessel head is unbolted and when the fuel is stored in the fuel pool, one train of the Standby Gas Treatment System shall be functional and aligned to the Reactor Building Zones of Units 1 and 2.	When the reactor vessel head is unbolted the capability of one train of the Standby Gas Treatment System to exhaust air shall be demonstrated weekly.
D. REACTOR BUILDING VENTILATION	D. REACTOR BUILDING VENTILATION
When the reactor vessel head is unbolted and when fuel is stored in the fuel pool the reactor building zones for Units 1 and 2 shall be ventilated such that a negative pressure is maintained with respect to the refueling zone.	When the reactor vessel head is unbolted and when fuel is stored in the fuel pool the reactor building zones for Units 1 and 2 shall be monitored daily to assure a negative pressure is being maintained with respect to the refueling zone.

3.7 CONTAINMENT SYSTEMS

E. INOPERABLE COMPONENTS

Whenever the requirements of specifications 3.7.B and 3.7.C cannot be met, all fuel handling activities or any activity over irradiated fuel in the vessel or fuel pool shall not be permitted.

3.7 BASES: CONTAINMENT SYSTEMS

A. PRIMARY CONTAINMENT

This specification ensures indication of adequate information regarding status of the drywell pressure and temperature and suppression chamber water level and temperature when fuel is in the reactor. When fuel is removed this requirement is no longer necessary. Monitoring of information concerning these primary containment parameters will ensure that sufficient control of these parameters can be manually initiated in a timely manner.

B. REFUEL ZONE VENTILATION

This specification ensures that an air flow is established in the refuel zone. Refuel zone ventilation may be accomplished by continuous operation of one refuel floor exhaust fan and one supply fan per unit. Fan dampers will be secured in the open position to ensure flow continuity.

Refuel zone air flow provides a means of exhausting moisture vapor in the unlikely event that spent fuel pool coolant steaming occurs.

C. STANDBY GAS TREATMENT

In the event that normal refuel zone ventilation is inoperable one standby gas treatment train is capable of providing a means of exhausting moisture vapor and air from the Reactor Building Zones of Units 1 and 2 to maintain a slight negative pressure between the reactor building zones and the refueling zone.

D. REACTOR BUILDING ZONES - Δp AND CHLORIDE CONTROL

The normal ventilation systems for the reactor building zones and the refuel zone are separate systems and will be aligned to maintain control of airborne chloride contamination from the reactor building zones to the refuel zone. Using this alignment, the reactor building zones will be maintained at a negative pressure with respect to the refuel zone and thereby preclude airborne chloride contamination of the refuel zone.

In the event of loss of off-site power the standby gas treatment system will perform the function of the reactor zone ventilation system.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
. Radioactive gases released from each units turbine and reactor building roof vents, the radwaste building roof vents, and the main stack shall be continuously monitored. To accomplish this, at least one reactor building and one turbine building vent monitoring system per unit shall be operating whenever that unit's building ventilation system is in service. Also one radwaste building system vent monitoring channel shall be operating whenever the radwaste building ventilation system is in service. At least one main stack monitoring channel shall be operating whenever any units air ejector, mechanical vacuum pump, or a standby gas treatment system train is in service. If normal monitoring systems are not available, temporary monitors or other systems shall be used to monitor effluent. A monitoring channel may be out of service for 4 hours for functional testing and calibration without providing a temporary monitor.	4.8.B Airborne Effluents
The primary containment shall be purged through the standby gas treatment system.	

C. Radiological Environmental Monitoring Program

An environmental monitoring program shall be conducted as described below and outlined in Table 4.8.F.

1. Atmospheric Monitoring

a. The atmospheric monitoring network is divided into three subgroups consisting of 12 monitoring stations of which 9 stations shall be operable at all times. The monitoring stations are shown on Figure 4.8-1. These monitoring locations are subject to change depending upon continued evaluation of the environmental monitoring program. The station at Muscle Shoals will be used as background reference.

Each monitor shall be capable of [illegible] sampling air at an [illegible] flow of approximately three cubic feet per

CONDITIONS OF OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the auxiliary electrical power system.

Objective

To assure an adequate supply of electrical power for those systems required for safety.

Specification

A. Auxiliary Electrical Equipment

For the condition of both units 1 and 2 in cold shutdown with fuel in the reactor, the following conditions apply except as specified in section 3.9.B.

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. Three diesel generators available and their associated 4 kV shutdown boards. All 4 kV shutdown boards energized unless offsite power is lost.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective

Verify the capability of the auxiliary electrical system.

Specification

A. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater. During the monthly generato test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers.

CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM
3. 4 kV shutdown bus 1 or 2 energized.	The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and speed shall be logged.
4. 480V shutdown boards associated with each unit are energized.	b. Once a month the quantity of diesel fuel available shall be logged.
5. Under voltage relays on the three 4 kV shutdown boards associated with the functional diesel generators shall be available.	c. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.
6. 250V DC shutdown board batteries and chargers for the four functional boards must be available.	d. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.
7. The 250V unit board batteries and a battery charger for each battery and associated battery boards must be functional.	2. D.C. Power System - Unit Batteries(250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt).
8. Batteries, chargers, busses, and feeds to the four SRM channels specified in Table 3.2.A shall be operable in each unit.	
9. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.	

CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

B. Cold Shutdown on Units 1 and 2 With Inoperable Equipment

1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.

2. When one of the three diesel generators becomes inoperable and a shutdown bus and three 4 kV shutdown boards are available, 2 days are allowed to comply with 3.9.A.2.

3. When one 4-kV shutdown board is inoperable, both off-site 161-kV transmission lines and both common station transformers shall be available and the remaining 4-kV shutdown boards and associated diesel generators, and all 480V emergency shutdown boards shall be functional.

4. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.

b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.

3. Undervoltage Relays

a. Once every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator w start.

b. The undervoltage relays which start the diesel

CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM
inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.	generators from the 4-kV shutdown boards, shall be calibrated annually for trip and reset and the measurements logged.
5. From and after the date that one of the four 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.	B. Operation with Inoperable Equipment 1. When one common station transformer, or one 161-kV line is found to be inoperable, all diesel generators and associated boards must be demonstrated functional immediately and daily thereafter. 2. When one of the three diesel generators is found to be inoperable, the remaining diesel generators and associated boards shall be demonstrated functional immediately and daily thereafter. 3. When one 4-kV shutdown board is found to be inoperable, all remaining 4-kV shutdown boards and two associated diesel generators, shall be demonstrated functional, immediately and daily thereafter.
C. Auxiliary Electrical Equipment For the condition of no fuel in Units 1 and 2 reactor vessels and irradiated fuel in the fuel pools for both Units 1 and 2 the following conditions apply except as specified in 3.9.D.	

CONDITIONS OF OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

1. Two off-site sources of power to the shutdown system available from either the 500 kV system (backfeeding) or the 161 kV transmission system.
2. Two diesel generators available and their associated 4 kV shutdown boards available for use.
3. 4 kV shutdown bus 1 or 2 energized.
4. The 480-Volt shutdown boards associated with the divisions of fuel pool cooling in service are energized.
5. Under voltage relays on the two 4 kV shutdown boards associated with the functional diesel generators shall be functionally operable.
6. 250-Volt DC shutdown board batteries and chargers for the two functional boards must be available.
7. The 250-Volt unit board batteries and a battery charger

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

C. Auxiliary Electrical Equipment

1. Diesel Generators

a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one-hour period at 75% of rated load or greater. During the monthly generato test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps sha be demonstrated, and the diesel starting time to rea rated voltage and speed shall be logged.

b. Once a month the quantity of diesel fuel available shall be logged.

c. Each diesel generator shall

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CONDITIONS OF OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM
and associated battery boards are functional.	be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.
8. There shall be a minimum of 103,300 gallons of diesel fuel in the standby diesel generator fuel tanks.	b. Once a month a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.
D. Fuel Removed from Units 1 and 2 Reactors and Stored in Units 1 and 2 Fuel Pools with Inoperable Equipment	2. D.C. Power System - Unit Batteries (250-Volt) Diesel Generator Batteries (125-Volt) and Shutdown Board Batteries (250-Volt).
1. From and after the date that off-site power is reduced to one source, another source must be restored within seven days.	a. Every week the specific gravity and the voltage of the pilot cell, and temperature of an adjacent cell and overall battery voltage shall be measured and logged.
2. When one of the two diesel generators becomes inoperable and a shutdown bus and two 4 kV shutdown boards are available, 2 days are allowed to comply with 3.9.C.2.	b. Every three months the measurements shall be made of voltage of each cell to nearest 0.1 volt, specific gravity of each cell, and temperature of every fifth cell. These measurements shall be logged.
3. From and after the date that one of the three 250-Volt unit batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during	

CONDITIONS OF OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

this period and the plans to return the failed component to an operable state.

4. From and after the date that one of the two 250-Volt shutdown board batteries and/or its associated battery board is found to be inoperable for any reason, the NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed component to an operable state.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

3. Undervoltage Relays

a. [illegible] every 6 months, the conditions under which the undervoltage relays are required shall be simulated with an undervoltage on each shutdown board to demonstrate that the associated diesel generator will start.

b. The undervoltage relays to start the diesel generator from the 4-kV shutdown bus shall be calibrated annual for trip and reset and the measurements logged.

D. Operation with Inoperable Equipment

1. When one common station transformer, or one 161 kV line is found to be inoperable, the t diesel generators and associat boards must be demonstrated functional immediately and daily thereafter.

2. When one of the two diesel generators is found to be inoperable, the remaining die

CONDITIONS OF OPERATION	SURVEILLANCE REQUIREMENTS
3.9 AUXILIARY ELECTRICAL SYSTEM	4.9 AUXILIARY ELECTRICAL SYSTEM generator and associated board shall be demonstrated functional immediately and daily thereafter.

3.9 BASES: AUXILIARY POWER SYSTEM

The auxiliary power system consists of 3 functionally operable diesel generators (A, B, and D) capable of automatic start on loss of voltage to their respective 4-kV shutdown boards. The C diesel feed is inoperable but its associated 4-kV shutdown board is available. The 480V shutdown boards 1A, 1B, 2A, and 2B are operable with at least one source of power. All the 480V reactor MOV boards 1A, 1B, 1C, 2A, 2B, and 2C are operable with at least one source of power.

A and B. Cold Shutdown on Units 1 and 2

The requirements of these specifications are based on the minimum complement of equipment to redundantly maintain the facility in a safe condition. The capacity requirements of this mode of operation can be adequately addressed with two diesels. The third diesel is to accommodate loss of off-site power and single failures in the diesel generator plant.

All 4-kV and 480V boards are available and the shutdown busses are available. With a single failure in any of the 4-kV and 480V board or feeders, diesel power can be directed to an adequate complement of equipment to accommodate the shutdown mode. This is accomplished by either feeding the required boards and their associated loads directly or by feeding two boards from one diesel through the backfeed capability of the shutdown bus.

There are eight 250-volt d-c battery systems each of which consists of a battery, battery charger, and distribution equipment. Three of these systems provide power for unit control functions, operative power for

unit motor loads, and alternative drive power for a 115-volt a-c unit preferred motor-generator set. One 250-volt d-c system provides power for common plant and transmission system control functions, drive power for a 115-volt a-c plant preferred motor-generator set, and emergency drive power for certain unit large motor loads. The four remaining systems deliver control power to the 4160-volt shutdown boards.

Each 250-volt d-c shutdown board control power supply can receive power from its own battery, battery charger, or from a spare charger. The chargers are powered from normal plant auxiliary power or from the standby diesel-driven generator system. Zero resistance short circuits between the control power supply and the shutdown board are cleared by fuses located in the respective control power supply. Each power supply is located in the reactor building near the shutdown board it supplies. Each battery is located in its own independently ventilated battery room.

The 250-volt d-c system is so arranged, and the batteries sized such, that the loss of any one unit battery will not prevent the safe cooldown of both units in the event of the loss of offsite power and single failure in any one unit. The loss of one 250-volt shutdown board battery affects normal control power only for the 4160-volt shutdown board which it supplies. The station battery supplies loads that are not essential for safe shutdown and cooldown of the nuclear system. This battery was not considered in the accident load calculations.

C and D. Fuel Removed from Units 1 and 2 Reactors and Stored in Units 1 And 2 Fuel Pools

The requirements of these specifications are based on the minimum complement of equipment to redundantly maintain the fuel in the fuel pool in a safe

condition. The capacity requirements of this mode of operation can be adequately addressed with one diesel. The second diesel is to accommodate loss of off-site power and single failures in the diesel generator plant.

Two 4-kV and two 480-volt boards will be available from off-site power supplied by one shutdown bus or from two diesel generators. With a single failure in any of the 4-kV and 480-volt boards or feeders, diesel power can be directed to an adequate complement of equipment to accommodate the fuel pool cooling mode.

The loss of any one unit battery will not prevent the safe cooling of both fuel pools in the event of the loss of offsite power and single failure in any one unit. The loss of one of the two 250-volt shutdown board batteries affects normal control power only for the 4160-volt shutdown board which it supplies.

4.9 BASES

The monthly tests of the diesel generators are primarily to check for failures and deterioration in the system since last use. The diesels will be loaded to at least 75 percent of rated power while engine and generator temperatures are stabilized (about one hour). The minimum 75 percent load will prevent soot formation in the cylinders and injection nozzles. Operation up to an equilibrium temperature ensures that there is no overheat problem. The tests also provide an engine and generator operating history to be compared with subsequent engine-generator test data to identify and to correct any mechanical or electrical deficiency before it can result in a system failure.

Battery maintenance with regard to the floating charge, equalizing charge, and electrolyte level will be based on the manufacturer's instruction and sound maintenance practices. In addition, written records will be maintained of the battery performance. The plant batteries will deteriorate with time but precipitous failure is unlikely. The type of surveillance called for in this specification is that which has been demonstrated through experience to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the Ampere-hour capacity of the battery, and will be applied as recommended.

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

Applicability

Applies to the removal of fuel from the reactor pressure vessel.

Objective

To prevent criticality during the removal of fuel from the reactor pressure vessel.

Specification

A. Reactivity Control

1. The reactor shall be kept in cold shutdown and all control rods fully inserted during fuel removal.
2. The reactor mode switch shall be locked in the "SHUTDOWN" position during fuel removal.

B. Core Monitoring

During fuel removal two channels of the SRM's, each on separate power supplies, shall be operable. For a SRM to be considered operable, the following conditions must be satisfied:

1. The SRM shall be fully inserted into the core.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

Applicability

Applies to the periodic verification of rod position and testing of instrumentation used during fuel removal.

Objective

To verify full insertion of control rods and operability of instrumentation during fuel removal.

Specification

A. Reactivity Control

1. Surveillance to verify full insertion of all control rods is specified in 4.3.
2. Prior to removing fuel, and daily thereafter, verify that the reactor mode switch is locked in the "SHUTDOWN" position.

B. Core Monitoring

Prior to the removal of fuel from the core the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

2. Prior to fuel removal the SRM shall have an initial minimum count rate of 3 cps with all rods fully inserted in the fully loaded core. The count rate will diminish during fuel removal.

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 8 1/2 feet above the top of the stored fuel.

2. Whenever irradiated fuel is in the spent fuel pool, the pool water temperature shall be ≤ 125°F.

3. Fuel pool water shall be maintained within the following limits:

Conductivity,
10 umho/cm @ 25°C
& Chloride, 0.5 ppm

D. Reactor Building Crane

1. The reactor building crane shall be operable:

a. When a spent fuel cask is handled.

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

response until the count rate diminishes to ≤ 0.3 cps.

C. Spent Fuel Pool Water Conditions

1. Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

2. Whenever irradiated fuel is stored in the spent fuel pool, the water temperature shall be recorded daily.

3. A sample of fuel pool water shall be analyzed in accordance with the following specifications:

a. At least daily for conductivity and chloride ion content.

b. At least once per 8 hour shift for conductivity and chloride ion content when the fuel pool cleanup system is inoperable.

D. Reactor Building Crane

1. The following operational checks and inspections shall be performed on the reactor building crane prior to

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

b. Whenever new or spent fuel is handled with the 5-ton hoist.

2. Whenever there is irradiated fuel in the spent fuel pool or in the reactor vessel with the reactor vessel head off, crane loads shall not be transported over the spent fuel pool or reactor vessel opening except those operations in the spent fuel pool and reactor vessel which normally require the services of the overhead crane.

3. When the crane is used in the unit 3 area and fuel is in the spent fuel pools or in the reactor vessel with the reactor vessel head off of unit 1 or unit 2, bridge movement to the unit 1 or 2 areas shall be restricted by redundant key lock electrical limit switches. Bypassing the limit switches for required activities in the units 1 and 2 areas shall be under

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

handling of a spent fuel cask and new or spent fuel (these need not be performed more frequently than quarterly):

a. The cab and pendant controls shall be demonstrated to be operable on both the 125-ton hoist and the 5-ton hoist.

b. A visual inspection shall be made to ensure structural integrity of the 125-ton hoist, the 5-ton hoist and cask yoke safety wire ropes.

c. The overtravel limit switch interlocks, movement speed control and braking operations for the bridge, trolley, and hoists, the pendant interlocks, the main-auxiliary hoist operation interlock, and the remote emergency stop shall be functionally tested.

d. The bridge drive limit switches shall be functionally tested each time they are placed in service before the crane is loaded.

3.10 CORE ALTERATIONS

strict administrative controls requiring the signature of a senior reactor operator.

4. When the crane is used to handle materials between units 1, 2 and 3 areas and irradiated fuel is in the fuel pools or in the reactor vessel with the reactor vessel head off of units 1 or 2, rail stops shall be installed to prevent trolley movement over the fuel pools or open reactor vessels. Removal of the rail stops to permit required activities specified in 3.10.D.2 shall be governed by strictiadministrative controls requiring the signature of a senior reactor operator.

5. When the crane is unattended for a period greater than one hour, the bridge and trolley shall be tied down to prevent movement in case of a natural phenomena.

4.10 CORE ALTERATIONS

LIMITING CONDITIONS FOR OPERATION

3.10 CORE ALTERATIONS

E. Spent Fuel Cask

1. Upon receipt, an empty fuel cask shall not be lifted until a visual inspection is made of the cask-lifting trunnions and fastening connection

F. Spent Fuel Cask Handling - Refueling Floor

1. Administrative control shall be exercised to limit the height the spent fuel cask is

SURVEILLANCE REQUIREMENTS

4.10 CORE ALTERATIONS

2. At any time when the reactor building crane is in service there shall be a person assigned to the refueling floor who has authority to control crane activities and will observe crane motions and conditions. He shall document at least once per shift that the requirements of Limiting Conditions for operation 3.10.D.2, 3.10.D.3, 3.10.D.4 and 3.10.D.5 have been complied with.

E. Spent Fuel Cask

1. Prior to attachment and lifting of an empty spent fuel cask from the shipping trailer, a visual inspection shall be conducted on the lifting trunnions and the fasteners used to connect the trunnion to the cask.

2. A visual inspection shall be made of the assembled trunnion on the empty cask to ensure proper assembly.

1.10 CORE ALTERATIONS

F. Spent Fuel Cask Handling - Refueling Floor

1. (Continued)

raised above the refueling floor by the reactor building crane to 6 inches, except for entry into the cask decontamination chamber where height above the floor will be approximately 3 feet.

2. The spent fuel cask yoke safety cables shall be properly positioned at all times except when the cask is in the decontamination chamber.

G. Refueling Interlocks

1. The fuel grapple hoist load switch shall be set at ≤1,000 lbs.

2. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load sensing limit switch on the hoist to be used shall be set at <400 lbs.

3.10/4.10 BASES: CORE ALTERATIONS

B. Core Monitoring

The SRM's are provided to monitor the core during shutdown and to monitor core reactivity and guide the operator while fuel is being removed from the reactor pressure vessel. All fuel moves during this operation will reduce reactivity. It is expected that the SRM's will drop below 3 cps before all of the fuel is unloaded. Since there will be no reactivity additions during this period, the low number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, SRM's will no longer be required.

Requiring the SRM's to be functionally tested prior to fuel removal assures that the SRM's will be operable at the start of fuel removal. The daily response check of the SRM's ensures their continued operability until the count rate diminishes due to fuel removal.

3.10/4.10 BASES: CORE ALTERATIONS

C. Spent Fuel Pool Water Conditions

The design of the spent fuel storage pool provides a storage location for approximately 140 percent of the full core load of fuel assemblies in the reactor building which ensures adequate shielding, cooling, and reactivity control of irradiated fuel. An analysis has been performed which shows that a water level at or in excess of eight and one-half feet over the top of the stored assemblies will provide shielding such that the maximum calculated radiological doses do not exceed the limits of 10 CFR 20. The normal water level provides 14-1/2 feet of additional water shielding. The capacity of the skimmer surge tanks is available to maintain the water level at its normal height for three days in the absence of additional water input from the condensate storage tanks. All penetrations of the fuel pool have been installed at such a height that their presence does not provide a possible drainage route that could lower the normal water level more than one-half foot.

The fuel pool cooling and clean up system can maintain the fuel pool water temperature below 125°F when removing maximum normal heat load from the pool with the RSCCW temperature at its maximum. If the pool temperature approaches 125°F the standby RHR loop through its fuel pool cooling connection will provide for continued heat removal.

The fuel pool water chemistry limits are established to prevent damage to the 304 stainless steel used to fabricate the fuel pool. Limits are placed on conductivity and chloride concentrations. Conductivity is limited because it is easily measured and gives

C. Spent Fuel Pool Water Conditions (Continued)

an indication of abnormal conditions and the presence of unusual materials in the water. The chloride limit is specified to prevent stress corrosion cracking of stainless steel and is based on established relationships between stress corrosion, chloride concentrations, and dissolved oxygen.

When conductivity is in its normal range, pH and chloride and other impurities affecting conductivity must also be within their normal range. Chloride measurements are made to determine whether or not they are also out of their normal operating values because conductivity could be high due to the presence of a neutral salt which would not have an effect on pH or chloride.

No additives are used in the fuel pool water and near neutral pH is maintained, therefore conductivity provides a very good measure of the quality of the water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the fuel pool, are exceeded.

Methods available for correcting off-standard conditions include using the fuel pool filter demineralizer, halting fuel or equipment movements, or using the condensate demineralizers. The samples of the fuel pool water which are taken daily are considered adequate to detect long-term changes in the chloride ion content. Fuel pool sampling frequency is increased to once per eight hour shift when the fuel pool clean up system is inoperable.

3.10/4.10 BASES: CORE ALTERATIONS

D. Reactor Building Crane

The reactor building crane and 125-ton hoist are required to be operable for handling of the spent fuel in the reactor building. The controls for the 125-ton hoist are located in the crane cab. The 5-ton has both cab and pendant controls.

A visual inspection of the load-bearing hoist wire rope assures detection of signs of distress or wear so that corrections can be promptly made if needed.

The testing of the various limits and interlocks assures their proper operation when the crane is used.

Assignment of a person to monitor crane activities and who has authority to direct crane operations assures close, safety monitoring and compliance with crane handling requirements. The placement of the bridge limit switches in their respective drive control circuits and the installation of the rail stops on the trolley rails assures that crane movements cannot be inadvertently made over spent fuel pools with fuel in them. Crane bridge and trolley tie down when the crane is unattended prevents their motion in the case of a natural event, such as an earthquake.

3.10/4.10 BASES: CORE ALTERATIONS

E. Spent Fuel Cask

The spent fuel cask design incorporates removable lifting trunnions. The visual inspection of the trunnions and fasteners prior to attachment to the cask assures that no visual damage has occurred during prior handling. The trunnions must be properly attached to the cask for lifting of the cask and the visual inspection assures correct installation.

F. Restoration Work

Requiring completion of fuel storage and installation of gates and blocks places the fuel in a maximum security condition prior to the restoration activities.

3.10/4.10 BASES: CORE ALTERATIONS

F. Spent Fuel Cask Handling - Refueling Floor

Although single failure protection has been provided in the design of the 125-ton hoist drum shaft, wire ropes, hook and lower block assembly on the reactor building crane, the limiting of lift height of a spent fuel cask controls the amount of energy available in a dropped cask accident when the cask is over the refueling floor.

An analysis has been made which shows that the floor and support members in the area of cask entry into the decontamination facility can satisfactorily sustain a dropped cask from a height of three feet.

The yoke safety cables provide single failure protection for the hook and lower block assembly and limit cask rotation. Cask rotation is necessary for decontamination and the safety ropes are removed during decontamination.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.11 UNIT 3/UNIT 1 AND 2 INTERACTIONS Applicability Applies to construction activities of unit 3. Objective To assure that unit 3 construction activities do not advercely affect the safety of either unit 1 or 2. Specification All operations, tests, and other activities associated with the construction of unit 3 that can affect any system or component required to be functional by these Interim Technical Specifications are prohibited until (1) an interation analysis is performed which shows that an activity associated with the construction and testing of unit 3 will not adversely affect the safety of units 1 and 2, and (2) the interaction analysis is approved by NRC.	

3.11 BASES: UNIT 3/UNIT 1 AND 2 INTERACTIONS

Since there are shared systems among Units 1, 2, and 3 an interaction analysis is performed which shows that an activity associated with construction of Unit 3 will not adversely affect the safety of unit 1 or unit 2.

TABLE 6.8.A

MINIMUM SHIFT CREW REQUIREMENTS

Shift Position	Present	During One Unit Fuel Unloading	During Two Unit Fuel Unloading	With Fuel in Storage Pool	Type of License
Shift Engineers	1	1	1	1	SRO
Assistant Shift Engineers	1	2	2	1	SRO
Licensed Reactor Operator [1]	1	1	1	1	RO
Unit Operators	2	2	2	2	RO
Assistant Unit Operators	6	6	7	4	NONE
Health Physics Technician	1	1	1	1	NONE
Other Personnel	2	2	2	0	--
Minimum Shift Crew	14	15	16	10	

NOTES: SRO - Senior Reactor Operator

RO - Reactor Operator

NOTE FOR TABLE 6.8.A

[1]This position is normally filled by an assistant shift engineer, but as a minimum it may be filled by a licensed reactor operator. When the incumbent is not a senior reactor operator, he shall not be assigned duties requiring him to direct licensed activities of reactor operators.

BASES

As a result of the damage created by the fire to various units 1 and 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located throughout the facility. Analyses have shown that in excess of 15 hours are available to perform these various tasks. In addition, studies have been performed to assess the amount of time

TABLE 6.8.A

BASES (Continued)

required to perform these manual operations including consideration of additional equipment failures. Results of these studies have shown that operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours. These studies were performed with the assumption of a shift crew of 10 men. We have required a shift crew of 14 men for the present cold, shutdown condition, and one additional man per unit during defueling operations, i.e., 15 men if one unit is being defueled and 16 men if both units are being defueled. The additional men will perform defueling operations and, since defueling will be suspended in the event a manual transfer to back-up systems is necessary, will provide margin for performance of the required manual operations and monitoring.

After defueling is completed with all the fuel in the storage pool, the time available to perform manual transfer to back-up systems is extended even further. Therefore, a shift complement of 10 men is then adequate to provide the same level of safety.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO THE FACILITY LICENSE DPR-33

AND AMENDMENT NO. 7 TO FACILITY LICENSE NO. DPR-52

(CHANGE NO. 11 TO TECHNICAL SPECIFICATIONS)

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR POWER PLANT, UNITS 1 & 2

DOCKET NOS. 50-259 AND 50-260

1.0 INTRODUCTION

On March 22, 1975, a fire at the Browns Ferry Nuclear Plant (BFNP) required the shutdown of Units 1 and 2. Since shortly after the start of the fire, both nuclear units have been in the cold shutdown condition with fuel in the cores and the respective reactor vessels vented to the containment atmosphere. Shutdown cooling systems are in operation in both units.

The fire affected various power, instrumentation and control cables which caused a number of emergency systems to become inoperable. Although many systems were damaged, the facility was placed and is being maintained in a safe and stable condition utilizing a substantial number of systems, and with functional backup systems being available. In fact, the total number of functional systems presently available for this purpose exceeds the number normally required for a facility in the cold shutdown condition.

The action now under consideration involves first steps toward the restoration of the facility. This involves the repair and removal of affected cables; however, before any repair and removal of cables and systems from the fire affected zone could be authorized, it was deemed prudent to require that all of the

necessary vital systems be protected, isolated or verified as free from interconnection with any of the systems or cables which were damaged by the fire. Therefore, on May 9, 1975, the Commission issued Amendment 9 to Facility License No. DPR-33 (Unit 1) and Amendment 6 to Facility License No. DPR-52 (Unit 2), Change No. 10 to the respective Technical Specifications (designated the Temporary Technical Specifications), and a supporting Safety Evaluation. This safety evaluation assessed the safety of the facility in its shutdown fire damaged condition, and the Temporary Technical Specifications set forth the requirements to assure that the units would remain in a safe and stable condition until fire damage removal and restoration could be subsequently authorized.

As authorized in the May 9, 1975, Amendments, the various vital systems and components required to assure the safety of the facility have been re-established as functional by (1) reconnecting them with electrical power and control cables routed outside of the fire damage zone, (2) verifying that the original cables were outside of the fire damage zone and therefore are unaffected by the fire, (3) converting some systems and components to manual control and (4) placing some systems and components in the desired safe configuration and disconnecting power leads thus preventing spurious operation. Additionally, the primary system mode switch is, by administrative control, locked in the "Shutdown" mode.

The original "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2," hereafter referred to as the "Plan", submitted by the licensee was based on leaving the cores in their respective vessels and isolating the required safety systems, to the extent possible, from potential interaction with fire affected systems during damage removal and restoration operations.

After discussions with the NRC staff, the licensee has determined that significantly greater safety would be provided by removing the fuel from both reactor vessels and placing it in the respective fuel storage pools prior to start of fire-damage removal and restoration operations. This has the effect of (1) virtually eliminating any potential for inadvertent criticality, (2) providing a substantially greater coolant inventory, and (3) reducing the number of systems required to maintain the facility in a safe condition. Thus, in the highly unlikely case where all fuel pool cooling is lost (heat removal would then be through boiling) and assuming no makeup, conservative analyses performed by the licensee have shown that a minimum of 160 hours are available before the pool level reaches the Technical Specification limit. The staff has verified by independent analyses that these estimates are very conservative. Thus more than adequate time is available for the restoration of pool cooling and makeup by manual operation of the available alternative systems. By comparison, approximately 28 hours would be available for this purpose before the core is uncovered if loss of cooling should occur with no makeup if the fuel were in the reactor vessel.

Accordingly, the licensee has requested authorization for removal of fuel from the cores and for its storage in the respective fuel storage pools,

and for removal of fire damage and restoration of the facility. A revised Safety Analysis (Section E of Part VI dated 5/28/75 and 6/5/75 of the Plan) has been submitted in support of this request. Therefore, this safety evaluation addresses the safety considerations involved in the removal of the fuel from the cores and its transfer to the respective fuel storage pools, and the subsequent storage of the fuel in the pools while fire damage removal is in progress. Change 11 to the Technical Specifications (designated the Interim Technical Specifications) are being issued concurrently with this safety evaluation as part of Amendments 10 and 7 to Facility Licenses Nos. DPR-33 and DPR-52 respectively. These Interim Technical Specifications were prepared and submitted for approval by the licensee after extensive discussions with the NRC staff. The Interim Technical Specifications (Change 11) include all the provisions for assuring the safety of the facility for the condition with fuel in the reactor vessels that were included in the Temporary Technical Specifications (Change 1). In addition, Change 11 adds the provisions for assuring safety during defueling, fuel storage, and fire damage removal and restoration.

The safety considerations discussed in this safety evaluation and the requirements of the Interim Technical Specifications will assure the safe storage of fuel while facility restoration work is in progress, as well as during fire damage removal work. However, we have not yet been provided with a complete

description of the design changes, criteria, and procedures for the repair and restoration of the facility. Therefore, this safety evaluation does not address and these amendments do not authorize the permanent repair and restoration of fire damage. Specific authorization in this regard will be issued after the necessary information has been submitted and reviewed and approved by the staff.

Likewise, this safety evaluation does not address and these amendments do not authorize the refueling and return to power operation of Browns Ferry Nuclear Plant Units 1 and 2. Any such authorization will be addressed in a subsequent safety evaluation and change to the Technical Specifications.

2. EVALUATION

2.1 General

This safety evaluatuion is based on a staff review of (1) the Safety Analysis (dated 5/28/75 and 6/5/75 included as Part VI, Section E, of the Plan; (2) Appendices A through G of the Safety Analysis; (3) other parts of the Plan as referenced in the Safety Analysis; (4) the BFNP FSAR; and (5) the information acquired through site inspections and extensive discussions with the licensee.

This safety evaluation addresses the safety considerations involved in the removal of fuel from the cores and its placement in the respective fuel storage pools (defueling), and the subsequent storage of the fuel in the pools while fire damage removal and restoration operations are in progress. This includes the defueling of Units 1 and 2, or the possible defueling of both units simultaneously. However, no cable cutting or fire damage removal operations will be permitted by the Interim Technical Specifications until the defueling of both units is completed, i.e., until all the fuel is in the respective fuel storage pool.

Safety is assured throughout the defueling-restoration operations by providing the systems and procedures required to accomplish the following:

a. Keep the fuel subcritical and monitor reactivity during fuel removal.

b. Remove decay heat from irradiated fuel in the reactor vessel and in the fuel storage pool and to maintain coolant over the fuel.

c. Assure the physical integrity of the fuel during fuel handling and storage.

d. Monitor and control the release of radioactive effluents to the environment.

e. Monitor and control the level of chloride contamination in the fuel pool.

Chloride contamination is not an immediate safety problem but it could contribute to a long term safety concern after power operation is resumed because of possible chloride stress corrosion effects on stainless steel components. However, careful control and limits on chloride contamination of fuel pool reactor coolant will help eliminate this long term concern.

An essential requirement for maintaining facility safety is the availability of adequate electric power. Therefore, the requirements for electric power is that the emergency power system shall retain delivery capability for the essential protection systems during all phases of the defueling-restoration operation following loss of all offsite power and failure of one diesel generator. The staff has reviewed the systems and inspected the facility to assure that this capability is available and will be provided by the Interim Technical Specifications.

The basic principle for assuring safety is that the minimum complement of systems and components be provided throughout the defueling-restoration operation as required to maintain BFNP Units 1 and 2 in a safe condition.
The Interim Technical Specifications require that this minimum redundancy be available at all times. However, it is expected that during most of the defueling-restoration operation the available safety systems will in fact exceed the minimum redundancy requirement, thus providing additional assurance for the health and safety of the public.

2.2 Reactivity Control

While fuel remains in the core it is essential to assure that control rods cannot be inadvertently withdrawn. To prevent this the reactor mode switch will be locked in the shutdown position, the directional control valves necessary for control rod withdrawal will be electrically disconnected, and the manual valve in the drive water supply to each hydraulic control unit will be closed. The Interim Technical Specifications include provisions to assure that this configuration is maintained thus making it virtually impossible to withdraw control rods. Also, before the reactor head is unbolted, full-in rod position indication and alarm must be operable in the control room. In this configuration the reactor core is subcritical and cannot be made critical by single failure or operator error.

In the highly unlikely event that a control rod drifts or becomes withdrawn, two channels of source range monitoring will be available to indicate an increase in count rate. The scram pilot valves and pneumatically controlled valves to scram each control rod will be operable. The control rod accumulators will be kept in a fully charged condition and a control rod drive pump will be in service. Manual scram is available to the operator and in addition, the source range monitors provide a non-coincidence input to the scram system This system will detect approach to criticality and will alarm and cause automatic scram action to drive in any control rod that is not in the full-in position.

The standby liquid control system, which can inject borated water to shutdown and maintain the core in a subcritical state, will be available while fuel is in the reactor vessel.

Once the fuel has been removed from the core and placed in the storage racks in the fuel pool, the physical configuration imposed by the racks prevents the attainment of criticality. In this condition, protection against inadvertent criticality is totally independent of any electrically operated system or component.

We have reviewed the measures and systems for reactivity control and monitoring and conclude that they provide a high degree of assurance that the facility will be protected against the occurrence of criticality throughout the defueling-restoration operation.

2.3 Residual Heat Removal Shutdown Cooling (Fuel in the Core)

Adequate cooling of any fuel in the core, either prior to or during the defueling operation, is provided by operation of one Residual Heat Removal (RHR) pump and associated heat exchanger. The decay heat (based on the entire core) which must be removed to prevent fuel damage is very small amounting to less than one-fourth of the heat exchanger capacity of one RHR system. Nonetheless, both the Temporary Technical Specifications (issued May 9, 1975) and the Interim Technical Specifications being issued with this safety evaluation require that two RHR loops with one pump per loop be operational, and three RHR pumps and heat exchangers are presently available for each unit.

The RHR pumps take suction from a recirculation loop and pump water

through the RHR heat exchangers in which reactor decay heat is transferred to a separate RHR service water system. In the event that the RHR pump providing flow should fail, the second RHR system for the unit is operational and can provide ample cooling.

In the event of the failure of the single valve in the suction line to the recirculation loop through which the RHR systems take suction, heat removal can be accomplished by the generation of steam in the core which is vented to the reactor building from the open reactor vessel (the head is off). Makeup coolant to the core in this mode of operation could be provided by either of the RHR pumps operating in the LPCI mode or the core spray pump taking suction from the torus. Operating in these modes the RHR systems do not rely on the single valve in the suction line to the recirculation loop.

In the cold shutdown head-off condition, with no pressure and low temperature in the primary system, there is no source to cause piping damage. Loss of coolant through inadvertent draining is minimized by locking valve positions so that water cannot be drained from the vessel and by providing double isolation valves in the operating cooling loop. However, even in the event of loss of coolant inventory, either of the RHR pumps or the core spray pump are capable of taking suction from the torus and providing water to the vessel.

With the reactor well flooded and the transfer canal gate open to the fuel storage pool the low decay heat level and the very large coolant inventory provide very ample time for manual switchover to backup equipment in the event of any operating cooling system failure. The time for such switchover was estimated in our May 9, 1975 Safety Evaluation Report to entail about 1 hour.

Analyses performed indicate that even if the refueling cavity were to accidentally drain, the reactor vessel volume is sufficient so that in excess of 28 hours would be available to restore coolant to the core before decay heat could cause sufficient boil off to reduce water level so as to uncover the top of fuel elements in the core.

We have concluded that sufficient decay heat removal and coolant makeup capability will be provided for the fuel in the core throughout the defueling operation, and that the fuel will thus be adequately protected from damage due to overheating.

2.4 Fuel Pool Cooling and Water Purification

Fuel cooling in the fuel storage pools for Units 1 and 2 is normally accomplished by the Fuel Pool Cooling and Purification System (FPPCPS). Each pool system consists of two fuel pool cooling pumps and heat exchangers arranged in parallel, two pool skimmer tanks, and the associated piping and valving. Each cooling pump and heat exchanger combination has sufficient capacity to remove the decay heat from the entire core.

The purification portion of the system is shared between Units 1 and 2 and consists of three precoat demineralizer trains arranged in parallel such that Unit 1 and Unit 2 use separate demineralizers with the third train maintained on standby to permit the backwashing and precoating of the exhausted demineralizer for either plant unit. Cooling is supplied by the two pumps and heat exchangers of the fuel pool cooling system.

When all fuel has been transferred to the fuel storage pool, one RHR system is available to function as a backup system in the event that all fuel pool cooling systems were lost, even though such loss could not be occasioned by any single event or failure. The Interim Technical Specifications require one RHR system to be operable as a backup for the fuel pool cooling systems, both of which are also required to be operable. One RHR pump and associated heat exchanger can provide ample cooling of the fuel pool, maintaining temperature below 125°F.

The licensee has upgraded the design of the spent fuel pool cooling system to Seismic Category I requirements. It was common for similar facilities licensed before Browns Ferry that the fuel pool cooling systems were not seismic Category I, although they were designed to prevent pool drainage in the event of failure (See Safety Guide 13). Nevertheless, at the time of the operating license review for Browns Ferry Unit 2, the licensee agreed to upgrade the fuel pool cooling system to seismic Category I. This involved installation of seismic Category I isolation valves and connections between the cooling systems and the purification system. The system is now seismic Category I and is designed to sustain normal loads, anticipated transients, the Operating Basis Earthquake, and the Safe Shutdown Earthquake within design limits which are consistent with those outlined in AEC Regulatory Guide 1.48, "Design Limits and Loading Conditions," for ASME Class 2 and 3 components. The specified design basis combinations of loading provide reasonable assurance that in the event (a) an earthquake should occur at the site, or (b) other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the spent fuel pool

cooling system components are not predicted to exceed the allowable design stress and strain limits for the materials of construction.

Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without loss of structural integrity. We conclude that the sytem design is acceptable and will perform as a seismic Category I system.

The licensee has analyzed various failure modes of the cooling system assuming a single initiating event followed by a concurrent single active failure. The failure modes include loss of offsite power, pump failures, valves failing in the unsafe condition and pipe cracks in the systems. For these failure modes, pool cooling will be available from at least one system. In addition, for highly degraded conditions where all pool cooling was assumed lost, pool boiling would occur and makeup to the pool would be available from three seismic Category I makeup systems; the standby coolant system connected to the pool through the RHR piping, and the two separated Emergency Equipment Cooling Water System (EECWS) headers each provided with a hose hydrant and hose to reach Unit 1 or Unit 2 pools.

We have reviewed the results of these analyses and agree with the licensee that for all postulated failure modes the stored fuel will be maintained at acceptable temperature levels and would not result in fuel damage.

Maintenance of the reactor coolant and the fuel storage pool water quality will be accomplished by the Reactor Water Cleanup System (when the fuel is in the reactor vessel) and by the fuel pool purification system when the reactor well

is flooded and after the entire core is in storage in the fuel pool. The licensee has analyzed the use of the purification system to maintain the Technical Specification limits and the capability of the radwaste system to process and store the effluents from the purification system. The results indicate that on the basis of one backwash and precoating operation a day per unit the radwaste system has excess capacity.

We have reviewed the radwaste capability using a conservative assumption of two backwash and precoating operations per unit per day. Under these conditions, the liquid waste water can be stored, processed and released to the environment at levels below those permitted for normal power operation of the plant. The resin sludge storage capacity is sufficient to store 3900 cubic feet of sludge. In addition, storage liners are on hand that can store 6200 cubic feet of dewatered sludge. Assuming no offsite shipment of resin sludge, adequate capacity exists for operation of the purification systems for approximately 6 months. This finding is based on no additional large discharges of radwaste to the system. Based on the above, we conclude that the fuel pool purification system, in combination with the radwaste system, can maintain the cooling water quality and not cause an increase in releases of radioactivity offsite above that already permitted for normal reactor operation.

2.5 Cooling Water Systems

The cooling water systems provided for Units 1 and 2 consist of the Residual Heat Removal Service Water System (RHRSWS), the Emergency Equipment Cooling Water System (EECWS), Raw Cooling Water System (RCWS) and the Reactor Building Closed Cooling Water System (RBCCWS). Each of these systems serves the function of removing heat from the reactor plant essential systems and

transferring it to the river (ultimate heat sink).

Since the fire, these systems have been made operational and have been valved into an arrangement that will assure cooling water to the RHR system, the spent fuel pool cooling system, the diesel generator cooling system, the standby coolant system and the manual fuel pool makeup water systems. Additional essential heat loads can be manually valved into the cooling water systems as required depending on the failure mode that occurs. For example, the EECWS can be manually valved to provide cooling water to the control room air conditioning coolers and emergency cooling units should the need arise.

These systems are redundant and separate and supplied with onsite power to assure a continued supply of cooling water for the essential systems. The Interim Technical Specifications require a minimum number of pumps that must be connected to the separate headers. The flow capacity of any one pump on any header is more than adequate to provide the cooling for the assigned essential systems.

The licensee has performed analyses for all potential failure modes and has demonstrated that a minimum of 8 hours are available to perform any manual operations to cool the fuel in the reactor vessel prior to boiling. When the fuel is in the storage pool and the gates are installed, a minimum of 30 hours are available for manual operations prior to pool boiling. The analysis also demonstrates that with one fuel pool pump and heat exchanger operating, the pool temperature can be maintained well below the boiling temperature. Two fuel pool cooling pumps or one RHR loop is required by the Interim Technical Specifications.

For the highly degraded case where all pool cooling is lost and the gates are in place, analyses demonstrate that, assuming no makeup, a minimum of 160 hours is available before the minimum pool level reaches the Technical Specification limit. This provides for more than adequate time for manual operations to provide makeup water to the pool from the EECW system or to restore pool cooling.

For the case where the refueling bellows seal were to fail during the fuel unloading from the reactor vessel, the analysis demonstrates that a minimum of 10 hours are available prior to boiling to provide makeup water to the fuel pool and assure cooling of the fuel in the reactor vessel.

We have evaluated the licensees analyses and the capability for cooling water systems to be manually aligned to the appropriate header to supply cooling water to the essential systems and to provide makeup water to the fuel pool or the reactor vessel in the event of the above failures, and have concluded that these systems are fully capable of supplying the vital cooling and makeup water needs of the facility throughout the period of defueling - restoration operations.

2.6 Electrical Power

The electrical energy to operate the systems required to maintain the units in a safe condition is provided by redundant offsite and onsite power supplies. The offsite power is supplied by four 161 kV circuits connected from the power grid to redundant shutdown buses which in turn supply power to all four 4.16 kV emergency buses. Three of the four plant diesels are capable of supplying emergency power on loss of offsite power. During fuel removal operations, the three diesels assure that emergency power is available for

sufficient cooling equipment to maintain a safe condition upon loss of offsite power plus the application of an additional single failure. During fuel storage and fire damage removal operations, only two diesels are required to assure sufficient cooling with the loss of offsite power and an additional single failure.

The diesels are cooled through the emergency equipment cooling water system whose pumps automatically start on diesel start to ensure sufficient coolant to the diesels to protect against overheating.

Redundant 250V batteries and their associated distribution systems are available to provide d-c control power as necessary to the required systems.

We have concluded that adequate power supplies are available to the systems and equipment required to maintain the facility in a safe condition, and that the provisions of the Interim Technical Specifications assure that this capability will be maintained throughout the defueling, fire damage removal and restoration operations.

2.7 Cable Cutting andDamage Removal

The safety of the facility during cable cutting and damage removal operations will be maintained by adherence to the following criteria and procedures.

The minimum systems required to maintain the facility in a safe condition have been identified. Prior to the start of any cable cutting or damage removal operations, the electrical cables serving these systems will be verified to be independent of the fire damage zone by careful check of the appropriate

electrical, schematics, installation drawings, and cable schedules. In some cases, this required replacement of fire damaged cables. Other measures being taken to preclude spurious operation of critical components include physically placing of some components in the required safe position, and the removal of power by opening breakers, pulling fuses and disconnecting wires. All disconnected wires will be taped to cover exposed conductors.

Also, prior to start of any cable cutting or fire damage removal, all fire affected cables will have been identified by careful check of the appropriate electrical schematics, and installation drawings and cable schedules. These cables will be disconnected from their terminations on both sides of the fire affected zone. Before a cable is cut, its disconnected ends will be checked for potential; if potential exists, no cutting will proceed until that cable is rechecked and cleared. Likewise, if potential is found at a cut end, the cable must be identified and cleared before further cutting.

Furthermore, if any of the critical structures, systems or components which are required to maintain the facility in a safe condition becomes inoperable, during cable cutting or damage removal, these operations will be suspended until the condition is rectified.

We have concluded that the measures and procedures governing cable cutting and fire damage removal operations described in Part VI of the Plan and the provisions of the Interim Technical Specifications provide adequate assurance that these operations can be performed without undue risk to the health and safety of the public.

2.8 Fuel Handling Equipment

The fuel handling equipment to be used for preparing for fuel removal and for transfer of the fuel from the reactor vessel to the fuel pool consists of the Reactor Building Crane (125 ton hoist and an auxiliary 5 ton hoist), the fuel handling bridge, and the normal complement of fuel handling tools, accessories and special tools that are required for normal refueling operations.

The reactor building crane will be used to remove the shield plugs, drywell head, reactor vessel head and the upper core internal to prepare for unloading the fuel from the reactor vessel. After the fuel has been transferred to the fuel pool, the crane will be used to secure the water tight integrity of the fuel pool, to replace the shield plugs in the dryer separator storage area and perform any required functions over the reactor well and empty reactor vessel.

The reactor building crane 125 ton hoist has been completely pre-operationally tested since the fire, including a 125% load test for the maximum expected load to be lifted. The 5 ton auxiliary hoist will also receive a similar test before use.

When the crane is not being used for the above described purposes, it will be placed in the area over the Unit 3 refueling zone and prevented from moving into the Unit 1 and 2 area by two functionally redundant and physically separated limit switches. The trolley movement will be similarly restricted from moving over the fuel pool (by rail stops installed on the trolley rails) except for the times when the 5 ton auxiliary hoist is

required to be used in Unit 1 and 2 for handling small loads in the fuel pool to accomplish required fuel handling operations. This exception also applies to performance of in-vessel inspection or work after the fuel is in storage. These exceptions will be under strict administrative control with signed approval required.

With these restrictions, limit devices and administrative procedures, we find there is reasonable assurance that heavy loads will not be carried over the fuel pool and thereby preclude damage to the fuel structural integrity or the stored fuel assemblies.

The fuel handling bridge, which removes the fuel from the reactor vessel and places it in the storage racks in the fuel pool, will be demonstrated to be operable prior to use. Testing to be performed will include the bridge drives, brakes, positioning drives, the mast and the fuel grapple. In addition, the lifting devices, including the 1/2 ton hoist, on the bridge will be load tested at 125% of their maximum expected load.

The only exception to the normal refueling safety features is the absence of the refueling interlocks that are associated with control rod drives and reactivity control. These interlocks will not be required for the current operation since the control rods are prevented from being moved by the deactivation of the drive system and by the operating mode switch being in the shutdown position which prohibits control rod withdrawal.

Based on our review, we conclude that the fuel handling and transfer equipment is capable of carrying out its function safely. Any malfunction of the equipment that could result in the dropping of a fuel assembly and cause a release of radioactivity has been analyzed in Section 2.10 and shown not to endanger the health and safety of the public.

2.9 Reactor Building and Control Bay Ventilation

Due to the effects of the fire, the reactor building ventilation system and the standby gas treatment system cannot be operated in their normal modes as required for normal power operation or accident conditions. As a result, the reactor building ventilation system is being manually operated to ventilate the refueling zone and other reactor building zones. One standby gas treatment system train will be verified to be free of the fire area and will be capable of being operated by onsite power. The system will be actuated manually at the standby gas treatment building.

With this arrangement, the reactor buidling will be ventilated such that the refueling zone will be at a positive pressure with respect to the fire affected area to control any chloride contamination of the refueling zone during the cable removal operations.

Since the reactor building ventilation system is not powered by onsite power it will not be available during a loss of offsite power condition. For this condition, the refueling zone will be maintained at essentially atmospheric pressure through dampers that have been secured in the open position; and the standby gas treatment train will be operated to provide a slightly negative pressure in the reactor building zones of Units 1 and 2.

The cooling of the RBCCW pumps and the spent fuel pool cooling pumps will normally be performed by the reactor building ventilation system. For the loss of offsite power condition, these pumps have been analyzed to be capable of continued operation for at least 24 hours assuming no heat loss from the building and no standby gas treatment system flow. In the event of an extended offsite power loss, these pumps would approach their design operating temperature limit and could be lost. In this case, the standby RHR fuel pool cooling system would maintain the pool cooling function since the RHR pumps have individual room coolers that are operable from onsite power and cooled by the cooling water systems previously evaluated in Section 2.5 of this report.

For the highly degraded condition where fuel pool boiling would occur, the vapor from the pool surface would be exhausted to the atmosphere through the dampers that have been secured in the open position, and through the ventilation tower doors to the roof of the reactor building. Any condensation on the inside of the reactor building siding will be collected by the refueling zone floor drain system.

We have reviewed the licensee's method for ventilation of the reactor building during normal and abnormal conditions. Based on our conclusions set forth in Section 2.10 of this report that secondary containment and standby gas treatment are not required to protect the health and safety of the public, we conclude that the proposed method of operation is acceptable.

The control bay ventilation and air conditioning systems are operable from

onsite power and will be available to maintain the habitability of the control bay and control the temperature of the essential areas, including the shutdown board rooms. The cooling function will be normally provided by the RCWS. However, as discussed in Section 2.5 Cooling Water Systems, the EECW system can be manually aligned to provide the cooling function if the RCWS is not available.

Based on our review, we conclude that adequate control bay ventilation and cooling can be provided for all postulated failure modes and that the proposed systems configurations are acceptable.

2.10 Accident Analysis

The principal accident event of concern in connection with the defueling and fuel storage operation is the potential for a fuel handling accident.

With low pressure and temperature and seismically qualified equipment, there is no potential for loss of coolant accidents. The analyses discussed in Sections 2.3, 2.4 and 2.5 above, demonstrate that the fuel will be amply protected from cladding damage due to lack of cooling.

The potential release of radioactivity resulting from pool boiling in the event of temporary loss of cooling has been considered. Even if all gap activity in the stored core were somehow released under this condition (an extremely unlikely event) the resultant offsite doses would be small fractions of the 10 CFR Part 100 exposure guidelines.

For the fuel handling accident, the licensee has analyzed the radiological effect of a postulated fuel handling accident (Appendix E of Part VI of

the Plan). We have evaluated the licensee's analysis and find it acceptable. However, we also performed an independent analysis and conclude that fuel handling at the Browns Ferry plant may be conducted without secondary containment and the Standby Gas Treatment System filters in operation without danger to the public health and safety. Although the licensee performed his analysis assuming only 46 days decay, the actual decay time before planned fuel transfer would be nearer 90 days. The analysis performed at 46 days results in a thyroid dose estimate which would be reduced by a factor of 16 at 88 days decay (32 days later) to a two-hour thyroid dose at the exclusion area boundary of less than 0.1 rem. These values are not only far below the siting guidelines of 10 CFR Part 100; they are, in fact, below the standards for radiation protection from a year's exposure to normal operational releases set forth in 10 CFR Part 20, for the general public. The whole body exposure is less than 0.1 rem at 46 days decay and not significantly less at 88 days. Table 2.10 gives the assumptions and results of the NRC staff analysis that support the conclusion that secondary containment and the standby gas treatment system are not required during fuel handling operations.

2.11 Process and Area Radiation Monitoring

Process and radiation monitoring equipment will be in service to monitor the following release points: (1) reactor turbine ventilation exhaust, (2) refueling zone exhaust, (3) radwaste ventilation exhaust, (4) turbine building roof ventilation, (5) liquid radwaste effluent, (6) main stack and (7) reactor zone exhaust.

In addition, area radiation monitors will be operable in both Units 1

TABLE 2.10 - ASSUMPTIONS AND CONSEQUENCES FOR A FUEL HANDLING ACCIDENT AT BROWNS FERRY

A. Assumptions

Power Level at Shutdown	3300 MWt
Operating Time	1000 days
Decay time from 3/22/75 to 6/17/75)	88 days
X/Q	3.4×10^{-4} sec/m^3
Breathing rate	3.47×10^{-4} m^3/sec
Number of rods in core	37,346
Number of rods breached	111
Iodine DF in water	0.01
Gap Activity	30% Kr-85 10% Other Noble gases and iodines
Power Peaking Factor	1.5
Puff release of all gap activity in breached rods assumed.	
Isotopes considered.	I-131, Kr-85, Xe-133

B. Consequences

Exclusion Area Boundary

Thyroid dose, rem	0.04
Whole Body dose, rem	0.04

and 2 reactor building and refueling floor areas for plant personnel safety.

This monitoring equipment provides assurance that plant releases are properly monitored and plant personnel are promptly warned of hazardous situations.

2.12 Reactor Coolant and Fuel Pool Cooling Water Quality

The licensee has described the precautions that will be employed to preclude the introduction of chloride contamination into the fuel pool cooling water and the reactor coolant. If chloride should be inadvertently introduced into the coolant, the licensee has shown that there are adequate means to detect and reduce the chloride to acceptable levels. The Interim Technical Specification limits on conductivity and chloride concentration and surveillance requirements provide reasonable assurance against conditions that could lead to significant stress corrosion of materials and loss of structural integrity of components.

2.13 Minimum Shift Crew Considerations

As a result of the damage created by the fire to various Unit 1 and Unit 2 cables, a number of safety systems cannot be operated from the control room. Equipment monitoring and operation is now required from various positions located through the facility. Analyses have shown that in excess of 28 hours are available before the fuel is uncovered to perform various tasks involved in transferring to the various functioning backup systems. Manual operation of equipment for which the most degradation is experienced could be achieved in a time interval of several hours, and the licensee

has indicated that these operations could easily be accomplished by a shift crew of 10 men.

We have required a shift crew of 14 men for the present cold shutdown condition, and one additional man per unit during defueling operations, i.e., 15 men if one unit is being defueled and 16 men if both units are being defueled. The additional men will perform defueling operations, and since defueling will be suspended in the event a manual transfer to backup safety systems is necessary, will provide margin for performance of the required manual operations and monitoring.

After defueling is completed with all the fuel in the storage pool, the time available to perform manual transfer to backup systems is extended even further. Therefore, a shift complement of 10 men is then adequate to provide the same level of safety.

These personnel will not be involved in any activities other than those required to maintain the facility in a safe condition. We have concluded that these staffing requirements provide a satisfactory level of safety with sufficient margin for anticipated contingincies and are acceptable.

2.14 Quality Assurance

As a result of our review and evaluation of the licensee's QA plan (Part XIII, Section A, dated May 28, 1975) for the restoration of BFNP Units 1 and 2, and as amended by letter dated June 11, 1975 from J. E. Gilleland to B. C. Rusche, we find it acceptable for the control of activities associated

with the defueling, fuel storage, and cleanup and removal of fire-affected equipment of BFNP Units 1 and 2. The QA plan, as amended, commits to the QA program requirements as described in TVA's Sequoyah Nuclear Plant FSAR (through Amendment 34), Docket No. 50-327, meets the requirements of Appendix B to 10 CFR Part 50, and meets, with some minor and acceptable exceptions, the requirements of WASH documents 1283, 1309 and 1284.

2.15 Unit 3 Construction Activities

To assure that Unit 3 activities do not adversely affect Unit 1 and 2, the technical specifications do not permit operations, tests, and other activities associated with the construction of Unit 3 that could affect the functioning of any safety system or component required by the Interim Technical Specifications until interface analyses are performed.

3.0 SUMMARY

"The Temporary Technical Specifications and Bases for Browns Ferry Nuclear Plant Unit 1 and 2, Change No. 10 to the Technical Specifications," issued May 9, 1975, included the appropriate specifications to provide assurance that the plant equipment and systems in their present status and configurations are adequate to provide the plant safety. The present status is that the reactors are in cold shutdown with the fuel in the reactor vessels.

In order to isolate the fuel to the maximum extent from possible adverse effects caused by activities associated with removal of fire damaged equipment or restoration work, the Unit 1 and Unit 2 fuel will be removed from the reactor vessels and stored in their respective fuel storage pools.

We have carefully reviewed the safety of the transfer of fuel to the storage pool and its continued storage there until completion of restoration work. The equipment and controls needed to perform the transfer of fuel (defueling) to the storage pool under the present conditions have not been affected by the fire (Ref. Sec. 2.8). The safety of the fuel is ensured by maintaining adequate cooling, with back-up cooling systems required to be operable, for fuel in the reactor vessel and in the storage pool during defueling-restoration operations (Ref. Sec. 2.3 and 2.4).

Maintaining the fuel in a shutdown condition is ensured in the reactor vessel and in the storage pool (Ref. Sec. 2.2). Redundant and diverse sources of make-up water are available in the event of loss of cooling or loss of coolant both in the reactor vessel and in the fuel storage pool (Ref. Sec. 2.3, 2.4, and 2.5).

Storing the fuel in the fuel storage pools provides a very large margin of safety. The present levels of decay heat and fission product inventory in the cores after approximately 90 days of decay are very low. The large coolant inventory provided by the fuel storage pools increases the time available to respond to the most extreme adverse events including the worst case of rupture of the refueling bellows seal that drains the coolant to within five inches of the top of the fuel and results in loss of the fuel pool cooling system (Ref. Sec. 2.5). Even in this case, at least 10 hours are available to take corrective action prior to the fuel storage pool water reaching the boiling temperature. This worst case is precluded after completion of defueling by the installation of the gate between the fuel pool and the reactor well. This is required by the Interim Technical Specifications.

The two pool cooling systems and the independent RHR back-up are both capable of providing ample cooling. These are also backed up by three independent make-up water systems to supply water to the pool to prevent the fuel from uncovering. Even if all cooling were lost it would take over seven days for the water level to drop to the Technical Specification limit, 8 1/2 feet above the fuel, if no make-up water is

assumed.

The activities and controls for removal of fire-damaged equipment have been reviewed and found to minimize any adverse effects on safety of the facility (Ref. Sec. 2.7). This is ensured by strict procedures, Quality Assurance requirements, and the isolation of the fuel in the storage pools from the restoration activities.

Therefore, Change No. 11 to the Technical Specifications includes the requirements for equipment and administrative controls needed in addition to those contained in the Temporary Technical Specifications, Change No 10, in order to perform the defueling and maintain the fuel in the storage pools with the protective systems and administrative controls described in this evaluation. The resulting Interim Technical Specifications continue to provide adequate protection to the health and safety of the public and, by accounting for the actual existing condition of the plant systems and components, they minimize any potentially unsafe conditions or actions.

The probability and consequences of any accident involving release of radioactivity from the site are very low, based on the following considerations: (1) The plant is locked into a shutdown condition prior to and during fuel removal from the reactor vessel; (2) The fuel will be stored in a non-critical geometry in the spent fuel pools that provide a large, inviolable of coolant with multiple sources of make-up; (3) The length of decay time since operation

results in a very low amount of decay heat generation from the irradiated fuel assemblies, as well as a low inventory of fission product activity for possible release; and (4) Redundant cooling systems and multiple makeup sources are available and a very long time period is available to restore cooling in the unlikely event of total loss of cooling systems. In view of the foregoing, we conclude that the issuance of these amendments does not involve a significant hazards consideration.

4.0 CONCLUSION

The changes in technical specifications authorized in connection with this evaluation result in enhancement of safety under present conditions, as discussed above. Based on these considerations, we have concluded that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 13, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 10 to Facility Operating License No. DPR-33 and Amendment No. 7 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments revise the Technical Specifications, taking into account the present condition of plant systems, so as to assure that the two units will remain in a safe and stable posture during the period of defueling of both units and storage of the fuel in their respective fuel pools. The plant will be maintained in this condition until completion of the repairs of damage resulting from the fire which occurred on March 22, 1975. These amendments also authorize the removal of fire-affected equipment. Approval of restoration of the facility will be the subject of a separate action.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules

and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated June 2, 1975, (2) Amendment No. 10 to License No. DPR-33 and Amendment No. 7 to License No. DPR-52 with Change No. 11, (3) Part VI Section E of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2", Revision 7 dated May 28, 1975, and Revision 10 dated June 5, 1975, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2), (3), and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 13th day of June 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT IV

LETTER FROM THE NRC TO THE TVA DATED SEPTEMBER 2, 1975, WITH ENCLOSURES

AMENDMENT NO. 14 TO DPR-33

AMENDMENT NO. 11 TO DPR-52

SAFETY EVALUATION

FEDERAL REGISTER NOTICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 2, 1975

Docket Nos. 50-259
and 50-260

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37201

Gentlemen:

Your letter dated August 29, 1975, requested approval of certain restoration activities for Browns Ferry Units 1 and 2 and revised the Final Safety Analysis Report for Units 1, 2, and 3. The restoration activities for Browns Ferry Units 1 and 2 and the details of the fire protection design changes for Unit 3 are described in "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20 (the Plan). In addition, your letter dated August 29, 1975, listed TVA's commitments for additional work not covered by the Plan.

We have reviewed the restoration work and design modifications proposed in the Plan and have considered the commitments made by TVA. We have concluded that TVA may proceed with the restoration and design modifications as proposed in the Plan. We find that implementation of these items are necessary and will not preclude any further modifications resulting from our continuing review, including resolution of the design details for incorporating the commitments made by TVA. Appropriate Amendments and Safety Evaluation are enclosed.

Sincerely,

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 14 to DPR-33
2. Amendment No. 11 to DPR-52
3. Safety Evaluation
4. Federal Register Notice

ccs: See next page

cc w/enclosures:
Robert H. Marquis
General Counsel
629 New Sprankle Building
Knoxville, Tennessee 37919

Athens Public Library
South and Forrest
Athens, Alabama 35611

Mr. Thomas Lee Hammons
Chairman, Limestone County Board
of Revenue
Athens, Alabama 35611

Anthony Z. Roisman, Esquire
Berlin, Roisman & Kessler
1712 N Street, NW
Washington, D.C. 20036

cc w/ enclosures & incoming
Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

Mr. Dave Hopkins
Environmental Protection Agency
1421 Peachtree Street, NE
Atlanta, Georgia 30309

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 29, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, add Paragraph 2.C(4) to Facility License No. DPR-33 to read as follows:

   "(4) The facility may be modified as described in Section X of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Date of Issuance: September 2, 1975

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 29, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, add Paragraph 2.C(5) to Facility License No. DPR-52 to read as follows:

   "(5) The facility may be modified as described in Section X of "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" dated April 13, 1975, and revisions thereto up to and including Revision 20."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Date of Issuance: September 2, 1975

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING SUPPORTING VARIOUS RESTORATION ACTIVITIES FOR THE BROWNS FERRY NUCLEAR POWER PLANT UNITS 1, 2 AND SUPPLEMENT 7 TO THE SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING U.S. NUCLEAR REGULATORY COMMISSION IN THE MATTER OF TENNESSEE VALLEY AUTHORITY, BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3 WITH REGARD TO DESIGN CHANGES

1.0 Introduction

On March 22, 1975, a fire at the Browns Ferry Nuclear Plant required a shut down of Units 1 and 2. The facility subsequent to the shutdown was found to have incurred substantial damage to power, control, and instrumentation wiring. Unit 3 of the Browns Ferry facility did not suffer any damage as a direct result of the fire. All three units are presently in the shutdown condition with the fuel removed from the vessels for Units 1 and 2; the Unit 3 reactor is still under construction with operation for that unit scheduled for early 1976. An overall program has been developed by the licensee delineating the necessary activities required to restore damaged portions of the facility to a level so that operation of Units 1 and 2 can be resumed.[1]

As a result of the fire an NRC review plan was developed consisting of three major and parallel elements. The first element was the investigation conducted by the Office of Inspection and Enforcement of events leading to the fire, fire fighting efforts, sequence of operational events and problems experienced with the nuclear steam supply system, interaction between units, and the response of TVA, State and Local authorities. This phase of the NRC plan has been completed.[2]

The second element of the plan, being performed by the Office of Nuclear Reactor Regulation has as its objectives (1) to assure that a safe plant configuration was attained and is being maintained subsequent to the fire, (2) to assure safety during removal of fuel from Units 1 and 2, (3) to assure plant safety during fire damage removal and restoration, and (4) to determine that the design changes that are required for restoration of these plants to operational status are acceptable. Thus far there have been two licensing actions taken to assure that the plant has been placed in a safe configuration following the fire. First, the plant Technical Specifications were changed to Temporary Technical Specifications designed to assure that a safe configuration

[1] "Plan for Evaluation, Repair and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975, Fire)" and revisions thereafter.

[2] OI&E Investigation Report of the March 22, 1975 Fire at the Browns Nuclear Station, dated July 25, 1975.

was maintained following the fire, while further corrective action plans were under development.[3] Subsequently, the Technical Specifications were again changed to provide for the removal of fuel from both Unit 1 and 2 reactor vessels and placement into the respective storage pools.[4] These changes also assure that the necessary minimum protective equipment to assure protection of the fuel in the storage pools from damage would be available with adequate redundancy and would be protected against adverse interaction with the fire affected wiring and cables, while such fire damage was removed and when the damaged wiring and cabling was replaced.

Thus far the restoration work performed at the facility to date has been limited to various clean-up operations and removal of damaged electrical wiring and cable trays. Thus, the fourth objective of the NRR element of the NRC program remains to be accomplished. This safety evaluation considers a portion of that objective, namely, the approval of certain proposed design features needed to prepare the facility for return to power operation. Consideration of return to operation including a determination of whether additional design features are required and an assessment of the operating procedures to assure safe operation will be discussed in a subsequent safety evaluation, prior to authorizing return to power operation.

The third element of the NRC program consists of the review being performed by the Special Browns Ferry Review Group established on March 26, 1975.[5] Efforts of this review group will be to prepare for NRC recommendations to change, as required, NRC policies, procedures, and technical requirements.

By letters dated August 25 and 29, 1975 TVA requested the approval of various restoration activities required to restore the plant to operational status as a result of the fire. The approval requested by TVA in the August 25

3 See Safety Evaluation issued in connection with Amendment 9 to license DPR-33 (Unit 1) and Amendment 6 to license DPR 1-52 (Unit 2) issued May 9, 1975.

4 See Safety Evaluation issued in connection with Amendment 10 to license DPR-33 (Unit 1) and Amendment 7 to license DPR-52 (Unit 2) issued.

5 Memorandum L Gossick to all NRC employees, "Appointment of Special Review Group," dated March 26, 1975.

letter consists of the following major items:

"1. Approval to proceed with structural work which covers restoration of concrete, embedments, pipe supports, cable tray supports and the replacement of pipe and ducting in the fire-affected area. In general, the plans related to this portion of the restoration effort are described in Part VIII of the Recovery Plan and in the response to Question 1.0 under the "General" heading.

2. Approval of electrical design changes including circuit changes and permission to proceed with the restoration and installation of cable trays, conduit, and cables, including approval for splicing. The work and design changes are described in Parts VI and X of the Restoration Plan.

3. Approval for installation of the fire detection system and for the installation of distribution piping of the fixed spray system in all areas of reactor building except in the fire-affected area. The installation in Unit 2 and Unit 3, and in the area outside the fire zone is scheduled to proceed simultaneously with structural work in the fire zone of Unit 1. These systems are described in subsection 5.0 of Part X of the Recovery Plan."

The replacement activities described in Item 1 above were authorized by letter dated August 28, 1975. The TVA letter dated August 29, 1975 modified Item 3 to some extent discussed below. Our evaluation of the overall restoration of the facility and improvement of facility design from the standpoint of fire protection is still under review. In its August 29, 1975 letter, the license agreed to provide further improvements in fire protection, over and above the improvements reflected in the changes authorized herein. These improvements are listed as commitments in the August 29, 1975 letter. The staff is still considering whether such further improvements are necessary prior to authorization of return to operation or whether such further improvements may be incorporated at a later shutdown period in the future. Although further improvements may be required in the future, including additional changes that may be required before return to operation is permitted, the design modifications authorized herein provide substantial enhancement of the capability of the facility to withstand the effects of a fire.

2.0 Description of Damage Resulting from the Fire

Substantial detail concerning the cause of the fire and the extent of damage to structures, systems and components is provided in the previously noted OI&E report. The following brief summary is provided as background for the discussion of the repair and replacement activities described in this Safety Evaluation.

After ignition, the fire propagated through the penetration in the wall between the cable spreading room and the Unit 1 reactor building. As a result of the pressure differential which is maintained beween the reactor building and the cable spreading room, and because an installed carbon dioxide (CO_2) fire extinguishing system was used in the cable spreading room, only a small amount of burning occurred in the cable spreading room. Damage to the cables in this area was limited to about 2 feet immediately adjacent to the penetration where the fire started. The major damage occurred in the reactor building outside the cable spreading room, in an area roughly 40 feet by 20 feet, where a very high concentration of electrical cables exists. There was very little other equipment in the fire affected area, with the only direct damage other than cables being the melting of a soldered joint on an air line.

Electrical System Damage

The electrical cables, after insulation had been burned off, shorted together, and grounded to their supporting trays or to the conduits, resulting in the loss of control power for much of the installed equipment such as valves, pumps, blowers and the like.

The most significant aspect of the fire damage was that it resulted in the loss of redundant safety equipment.

Soot and Chloride Damage

In addition to the cable damage, the burning insulation created a dense soot which was deposited throughout the Unit 1 reactor building and in some small areas in the Unit 2 reactor building. An estimated 4,000 lbs of polyvinyl chloride insulated cable which burned released an estimated 1400 lbs of chloride to the reactor building. Chlorides are very corrosive to most materials. Components subject to stress corrosion damage from chloride contamination are limited to those

constructed of stainless steel and high strength alloy steel such as equipment hanger springs. The licensee has provided a program of cleaning, cleanliness verification, evaluation, and surveillance to mitigate any harmful consequences of the exposure to chlorides.

TVA's cleaning program consisted of dry cleaning and wet detergent cleaning using demineralized water and a chloride-free industrial detergent solution. Cleaning procedures were repeated until measured chloride levels were below 0.08 mg per dm^2. This level of chloride contamination is recognized to be an acceptable level for chloride in contact with stainless steel and is incorporated as the acceptance level in RDT Standard E5-1T. After cleanliness was verified, precautions were taken to prevent recontamination. All sources of contamination were removed and further surveys were performed to verify acceptable cleanliness levels.

In addition to cleaning efforts, the licensee will undertake an extensive program, consisting of liquid penetrant examination, metallurgical sampling and surface replica examination. During the penetrant work, the stainless steel piping and piping components will be tested by liquid penetrant in all of the principle piping areas outside of primary containment in which corrosion damage could have occurred.

Metallurgical samples will be taken from stainless steel piping, stainless components and high strength alloy steel components. The samples will be removed in the fire zone and other areas of the Unit 1 reactor building where heavy residue from the fire was deposited. In addition, surface replicas will be taken from Unit 1 piping. Surface replication is a very sensitive technique with respect to the detection of cracking due to chloride stress corrosion and was an addition to the original metallurgical evaluation plan.

The surveillance program will be worked out in detail between TVA and NRC and will be carried out before return to operation is authorized and thereafter during plant operation, and will be addressed in our subsequent safety evaluation before return to operation is authorized. However, the initial proposed program consists of quarterly visual examinations of all stainless steel components and piping for signs of leakage, cracking or distress.

We conclude that based on our evaluation the cleaning procedures were effective in reducing chloride levels to or below the recognized acceptance standards. Testing to be carried out during the restoration period along with the subsequent surveillance program which will be developed before the plant returns to operation will assure that any deleterious corrosion will be detected and corrected in a timely manner.

3.0 Overall Approach Proposed by TVA

The Browns Ferry Nuclear Facility consists of three boiling water nuclear reactors each of which are designed to produce 1067 Mw of electric power. Units 1 and 2 of that facility were authorized for operation in June 1973 and June 1974 respectively and were in operation up until the March 22, 1975, fire. Unit 3 is still under construction and is expected to go into service early in 1976.

Units 1 and 2 share a common control room with a shared cable spreading room located beneath the control room. Wiring carrying signals between the control room and various pieces of equipment in the plant are routed into the cable trays in the cable spreading room. The fire caused extensive damage to wiring located immediately outside of the cable spreading room.

Some sharing of equipment exists between all three units; the sharing of electrical systems is most extensive in Units 1 and 2. The electrical design concept employed in the design of the Units 1 and 2 is based on using a two-division concept. The purpose of dividing the electrical system into two divisions was to assure that the facility could be maintained in a safe configuration even with the postulated loss of one entire division.

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Consideration of a fire as a design basis event, requires that the capability be maintained during and following the fire to safely shutdown the reactor and remove the decay heat. Previous analyses (PSAR and FSAR) have shown that only one division of electrical equipment is required to assure that the facility is maintained in a safe condition. The control rod system in a BWR is designed to be fail safe. That is, a loss of electrical power results in initiating a scram of the control rods and shutdown of the reactor.* Therefore, the major objective for protection of the reactor from the effects of fire is to assure that necessary equipment to remove decay heat remains operable in the event of a fire.

Thus, it is important to limit fire induced failures to one division of the safety related equipment for shutdown heat removal from each reactor which provides, for each reactor at least one core spray system with two associated pumps and valves, and one RHR system with associated pumps, heat exchangers, and valves, together with enough relief valves for reactor system blowdown to low pressure. (It should be noted that non-safety grade equipment is normally used to perform this function). In order to assure that a fire would not cause a loss of both divisions, some changes in the facility design and equipment layout are necessary.

Facility modification that would be required to allow complete physical separation of the divisions would be extremely difficult to achieve in an already designed and existing facility. There are however, combinations of methods which can be used to accomplish the same purpose, e.g., 1) fabrication of the electrical distribution system with fireproof wiring, 2) fire detection and extinguishing systems that protect the critical areas of the plant, 3) elimination of substances which can be the source of fire would also reduce the potential for fire damage to both electrical divisions; and 4) relocation of the divisions as well as installation of physical barriers to limit the susceptibility of damage occurring in both divisions.

In recognition of the difficulty of achieving complete physical independence of the two divisions, the licensee proposed an overall program that con-

* The control rod system in a BWR is a mechanical system whose operating power is derived from a hydraulic system. An interruption of the electrical power to the scram circuitry causes a scram which initiates the hydraulic system and becomes independent of the electrical system and eliminates the need for continuous power supply.

sidered a number of elements. These include design changes, improved fire fighting methods, and improved administrative procedures which would assure that postulated fires within the facility would not result in a loss of both electrical divisions. This overall approach proposed by the licensee is presented in Part X of the recovery plan. The essential ingredients of that proposal are to provide changes to the electrical circuits via either sufficient separation or by the installation of various fire barriers to assure that a fire induced failure in one division could not damage the second division. A one-hour fire protection interval was selected to assure that a fire in any location in one division would not damage any portion of its redundant division even if the fire should last for one full hour. The licensee analyzed its systems taking into account the materials pertaining to the particular arrangement of the division of cabling under consideration to determine the need for additional protection. Using these analytical techniques, fire barriers are proposed to be installed such that the stated objective of a one hour fire protection interval would be available through the facility. On the basis of a one hour fire protection interval, procedures were derived to assure that a fire which could threaten the safe shutdown of the facility, would be extinguished well within one hour. Reliance is placed on using augmented administrative procedures in conjunction with hand held fire protection equipment already installed in the facility and additional hand held extinguishers which will be added throughout the facility. In those areas for which easy access by personnel with hand held fire equipment would be difficult, licensee will install fixed manually operated water sprays. The areas which have been specifically identified as requiring fixed spray systems are as follows:

a. Along cable tray runs parallel to the north wall of Reactor Building at elevation 593 feet of units 1, 2, and 3.

b. All penetrations from the cable spreading room into the turbine building and from the cable spreading room into the Reactor building containing congested cable trays where several trays make each inaccessible.

c. In other Reactor building areas where concentration of cables, as determined by visual surveys, make it difficult for impediate, easy access for fire fighting application of extinguishing agents.

To assure that any fires that may start are promptly detected, so that they may be extinguished promptly, the licensee has proposed installing a substantially extended fire detection system which will detect the start of fires and will promptly alarm in the control room. Operating procedures under development will result in prompt dispatch of fire fighting personnel to extinguish the fire.

Two types of detector will be employed to provide the fire detection function in those areas determined from the plant reanalysis to be critical. Products of combustion detectors will be installed in general on a 30 ft. grid basis. These areas will be zoned and no less than two detectors will be installed in a given zone. All detectors within a given zone will be wired in "OR" logic. Sensitivity or trip point will be 6 milligrams of products of combustion per cubic foot of air. Heat cable detectors will be installed on cable trays in the critical areas. This will provide both redundancy and diversity of detectors. The heat cable detectors will respond when the temperature at any point along the cable reaches 250 F. All sensors will be tested on a periodic basis. Products of combustion detectors within a given zone will provide one alarm for detection and another alarm for circuit trouble. Heat detectors within a given zone will provide one alarm for detection of heat or loss of power. And all detectors will receive power from the plant preferred bus.

Cable Replacement and Repair

The major effort in restoring the facility to a condition in which it will be able to operate, is the replacement and repair of damaged wiring and cables. This amounts to some 9500 conductors to be replaced or spliced. All cabling of the reactor protection system, primary containment isolation system, engineered safeguards (divisional) cables will be replaced from terminal to terminal (without splicing). Only non-divisional cables (that is those not required for providing either power or control to safety systems) will be spliced and repaired. Of these, the specific repair requirements provide for protection against high temperature or interruption of vital circuits that could result from a poor splice. The procedures proposed call for splicing at undamaged locations determined by measurements made on the insulation material, for splices in accordance with applicable codes; for accessibility to facilitate inspection; and for measures that assure no mechanical loading exists on the splices. Appropriate fire stops are provided along the cable trays away from the splice.

Replacement or repair of cabling will be of equivalent quality or better than the original cabling which was damaged by the fire.

Circuitry Changes

In addition to replacement and repair of damaged wiring and cables, TVA proposes design changes to enhance fire protection and eliminate the source of the loss of redundant safety equipment which occurred as a result of the March 22, 1975 fire.

The most significant loss of redundant equipment was associated with failures of their power sources which were caused by short circuits to lamp circuits leading from the control circuits of Reactor Motor Operated Valve (MOV) boards. These cables were considered to be non-divisional because a dropping resistor was provided in the lamp circuit for the purpose of isolating this circuit from the control circuit. Several of these cables from both divisions were run together in common trays and were, subsequently, all damaged by the fire. TVA has proposed to remove the cables leading from Reactor MOV board control circuits to breaker indication lamps to eliminate the major problem which was the loss of ability to operate the boards in both divisions. We agree with this approach and conclude that these circuit changes will eliminate the loss of redundant division MOV board controls.

The second most significant cause for loss of redundant equipment was due to the proximity of conduits containing division I and divison II cables to cable trays which were the primary source of combustible material. Although the use of conduit may provide adequate protection against some affects of fire, the fire which occurred proved that conduit protection alone was insufficient for a fire of that magnitude without additional protection in the form of additional separation or fire barriers. To remedy this TVA will modify the cabling and wiring systems to assure that cross divisional affects during fires due to the proximity of conduits are minimized. To accomplish this the cabling and wiring is to be modified, wherever divisional cables are routed in conduit near open trays carrying cables of the opposite division. Separation of divisional conduits will be provided by structures, distance, barriers, interposing ducts, pipes, etc., or combinations thereof to provide at least a one hour fire protection interval. The proposed criteria to be used to effect these changes are contained in Part X of the plan.

An analysis of the plant cable tray and conduit network has been made using the proposed criteria and approximately 80 locations have been identified for which conformance to the above proposed requirements had to be examined. The supplemental separations requirement involves the use of barriers and fire stops to inhibit cross divisional effects during a fire and to minimize fire propagation within divisions. In addition, conduits are no longer treated as adequate barriers to fire and therefore separation or other barriers are required to protect the conduit circuits. Together these additional requirements will provide a minimum of one hour fire protection interval against cross divisional effects.

A third cause for loss of redundant equipment was damage to their cables sharing the same cable tray contrary to the stated criteria. This third significant cause for loss of equipment will be remedied by separating the involved cables. Selective changes will also be made in the electric power system to improve isolation. These changes will provide individual normal feeders to the 4KV/480-V transformers. One change will eliminate sharing between units of the 4 KV feeder that was the normal supply to 480 V Shutdown Board 2A and 2B through TS3E. Other changes will provide individual 4 KV feeds to transformers TDA and TDB which supply 480-V Diesel Auxiliary Boards A and B, respectively. The only shared 4KV feeder will be the alternate supply to transformers TSIE and TDE.

The requirements for replacement and for splicing and repair of cables damaged by the fire will provide for replacement in which the cabling integrity and performance will be essentially equivalent to that provided in the original installation. The design changes in circuitry will provide substantially enhanced protection for the redundant safety system circuitry by providing increased separation so as to provide at least a one hour fire protection barrier to assure that fires in one division will not adversely affect another division even if the fire should last for a full hour.

With the increased and improved fire detection system proposed and the installation of fixed sprays in areas in which access is restricted to augment improved fire fighting procedures throughout the plant, the design changes which TVA proposes will substantially improve

the ability of the facility to withstand fire.

4.0 NRR Evaluation of the TVA Approach

Although the fire that occurred caused damage to the electrical system greater than that which had been considered in our original evaluation of that facility, our evaluation has shown that even with this extensive damage considerable flexibility remained with respect to methods available to remove the decay heat from the reactor and assure the reactor was maintained in a safe condition.

In spite of the capability already inherent in the facility, the fire demonstrates that additional emphasis on the protection of the facility from a fire is needed. In this connection we have considered those features, actions and design approaches that should be a part of fire protection capability The three basic considerations in minimizing the effects of the fire are: administrative actions that can prevent a fire from occurring; use of separation as a mechanism by which to prevent a fire from damaging redundant safety equipment; and, incorporation of a means to detect and extinguish a fire quickly.

The principal difference in our approach from that used by TVA, is that we would emphasize that each of these elements should be considered by themselves to the extent practical. As previously noted, no one of these three elements can be relied upon to completely eliminate problems associated with fires. TVA's approach instead was based on selection of a one hour fire protection separation and providing detection and fire protection equipment and procedures to extinguish fires within one hour. This is to be combined with strict administative control to prevent fires from occurring.

As indicated above the one hour fire protection separation which the changes in circuitry will provide, will substantially enhance the ability of the facility to withstand fire. However, we believe and have indicated to TVA our position that such separation should be considered as providing a base protection standard and that if practicable application of additional thermal barriers could significantly extend the one hour interval, then such additional barriers should be installed.
TVA will substantially upgrade its operating procedures and training for fire protection to assure that fires can be extinguished within one hour using the hand held fire equipment available in the facility. TVA proposes to install a fixed manually operated spray system, to supplement the hand held equipment, only in areas in which access prevents effective use of hand held fire fighting equipment.

We believe that an automatically actuated fixed spray system should be extended throughout all areas in which the potential for inter-divisional effects exist. The advice we have received from expert consultants supports this approach. The experience gained as a result of the Browns Ferry Fire indicates that if a water spray had been used earlier in the fire, considerably less damage would have resulted and the fire would have been extinguished sooner than it was. After discussions between the staff and TVA, the licensee in a letter dated August 29, 1975, has committed to converting those fixed spray systems he has presently designed to an automatic system within approximately one year. He has also committed in that letter to evaluate the installation of additional fixed spray systems to extend the fire protection coverage throughout all those areas in which such protection is required. Such studies will be completed and any areas for which this extended coverage should be provided will be identified and discussed in our subsequent evaluation prior to authorization of power operation of any of these units.

The operation of the installed CO2 (Carbon Dioxide) fire extinguishing system in the cable spreading room was effective in preventing substantial damage to the cables located in that room. The licensee did not propose to make any further changes to the fire extinguishing equipment provided in the cable spreading room. Reliance on the manual operation of CO_2 system as is now provided is not adequate. We have informed TVA of our conclusion and TVA has agreed to make necessary changes to provide automatic actuation of the CO_2 system. We have also informed TVA that we believe that there should be a liberal application of a flamemastic material to coat the cable trays located within the cable spreading room. The CO_2 system when exhausted may not have sufficient cooling capability to fully extinguish a deep-seated fire. The licensee has indicated that in such an event the system could be augmented to extinguish any fire by using hand held equipment. Although this may be an appropriate method for augmenting the CO_2 system, we have indicated our position to the licensee that he should study further additional measures, including consideration of installing fixed water spray systems, that could be used within the cable spreading room. The licensee has agreed to provide such a study. The study will be completed and any changes will be identifed and discussed in our subsequent safety evaluation prior to authorization of return to power.

We have informed the licensee that the design of the penetration which requires building a fire stop be modified to incorporate a new material which will be demonstrated by tests to be of a fire resistant nature. This would include all penetrations that were damaged as a result of the fire as well as all new penetrations. Those in the Unit 3 cable spreading room are to be constructed using the new design. We have also informed the licensee to reexamine the penetration that now exists and assure that they are placed into the originally designed condition. We have further asked that whenever it is necessary as a result of the restoration program to breach a fire stop that the repair of that fire stop to the extent practical include removal of the original materials with replacement made of materials used for the design of new fire stops. The licensee has agreed to these provisions.

During our detailed evaluation we and our consultants have looked into the manner in which the ventilation systems throughout the plant should perform. We have concluded that in the event a fire is detected in the cable spreading room, its ventilation system should be stopped. We have not been able to complete our determinations as to whether the ventilation system used for other rooms should also be stopped in the event of a fire. There is a need for ventilation to clear and remove the combustion products thereby providing accessibility into the fire area. On the other hand, the increased ventilation rate may be undesirable with respect to the capability for causing the fire to burn with an increased force. We have informed the licensee that this matter must be studied further prior to authorizing any power operation. The licensee has committed to perform this further evaluation.

Our evaluation of the facility will continue and our subsequent evaluation prior to facility operation will discuss further facility improvements still under study as described above. However, authorization of the changes and the work activities proposed by TVA will improve facility safety as described in Section 5.0. Moreover, such work is of such a nature that further changes to effectuate potential improvements discussed above would not be precluded or hampered by the restoration activities approved at this time.

5.0 Requested Approval

There are three major action items for which the licensee has requested approval, namely, structural work, electricl design changes including installation of cables, and installation of the fire detection and spray systems. Following are the bases for the approval for each of these action items.

1. Approval to proceed with structural work which covers restoration of concrete, embedments, pipe supports, cable tray supports and the replacement of pipe and ducting in the fire-affected area. In general, the plans related to this portion of the restoration effort are described in Part VIII of the Recovery Plan and in the responses to Question 1.0 under the "General" heading.

Basis for NRC Approval

The programs and procedures for restoration of process piping, HVAC ducts and supports, hangers, restraints, cable trays and supports of affected mechanical equipment have been reviewed. The restoration program requires the evaluation of all the above mentioned components for the effects of the cable fire. Thermal sensitivity based on minimum acceptable material properties and safety function will be determined for these components and items and then compared to the fire temperature zones determined by inspection of damage and analyses. Where thermal sensitivity was lower than the actual temperature experienced, the component or item may be eliminated from further consideration and replaced to the original specification or an evaluation will be made to justify continued use of the component.

Evaluation of all process piping is required regardless of the level of exposure to temperature effects. In lieu of further evaluation some piping may be replaced to the original specifications. If piping is to remain in service the restoration program requires removal of insulation, complete cleaning of the pipe surfaces and dye penetrant inspection of the pipe and pipe fitting surfaces.

The scope of the dye penetrant inspection is set in accord with the piping material strengths and sensitivity to the chemical environment resulting from the fire.

Carbon steel piping found to have been in a 500 F temperature zone will receive an initial inspection of twenty percent of the pipe surface and one

hundred percent of the fittings. The discovery of fire related cracks will require inspection of one hundred percent of the pipe surface in the same temperature zone. The boundary of the temperature zone will be extended by ten feet if the crack is found within ten feet of the initially determined boundary.

Stainless steel pipe and fittings will receive a one hundred percent inspection.

Aluminum piping will receive an inspection similar to that specified for carbon steel with the exception that the temperature zone is reduced to below 340 F. Any aluminum pipe found exposed to temperatures above 340 F will be replaced without further evaluation. In addition, if any fire related cracks are found within twenty feet of the 340 F zone a one hundred percent inspection will be performed in that twenty foot length and inspection will continue for another twenty feet of length.

We find that these procedures for restoration of process piping are properly set in accord with the material properties of the piping material and in recognition of the sensitivities of the various materials to the chemical environment resulting from the fire and that these procedures provide adequate assurance that the restored process piping will be fully capable of performing all design basis safety functions.

Safety related heating ventilating and air conditioning ducts will be evaluated on the basis of exposure to temperature of 500 F or greater as gaged by the established temperature zones and visible distortion of the ducts. All ducts showing visual distortions will be replaced or reworked to original specifications. All duct work influenced by the fire will be cleaned regardless of temperature exposure.

Evaluation of cable trays, cable tray supports and fixed members of pipe supports will be based on exposure temperatures and stress analysis considering material strengths reduced by any possible annealing affects of higher temperature exposure. Any structural steel found to have been immersed in a temperature of 1000 F or higher will be replaced without further evaluation.

In general all cable trays in direct and near contact with cables actually consumed by the fire will be replaced without further evaluation. The larger cable tray supports will be re-evaluated based on the actual load to be sustained. The acceptable stress levels for these analyses will be set at either seventy five percent of the original design allowable stress or based on the measurement of actual physical properties from a sample of the material exposed to the actual fire conditions.

The requirements specified above for acceptance of cable tray supports will also be employed for acceptance of the fixed members of pipe supports. Additional requirements have been established for acceptance of the variable elements of pipe supports. In the absence of variations in setting from the effects of the fire and the absence of actual fire damage the support setting will be corrected to give a properly aligned piping system. Setting changes as a result of the fire with no visible evidence of heat damage will require evaluation of spring characteristics to assure the necessary range of support as required by the system design. Visible heat damage will require replacement of the spring mechanism and if necessary the entire support.
We find that the procedures set forth for the evaluation and restoration of the safety related HVAC ducts cable trays, cable tray supports, and the fixed and variable members of pipe supports provide adequate assurance that these items will be capable of performing as designed throughout the range of loads and functions specified in the design bases for Browns Ferry Unit 1.

With respect to the structural steel components and the reinforced concrete structures which either support the cables or were in the vicinity of the burning cables TVA has initiated a restoration program which consists of:

1. Identification of the affected structures;
2. Establishment of the most probable temperature to which each structure was subjected during the fire;
3. Evaluation of the extent of damage of each structure;
4. Establishment of criteria for the requirement of replacement or repair of the damaged structure; and,
5. Development of procedures for detailed repair.

The structural steel components affected consist of miscellaneous structural steel supports, steel embedment, and portions of the building superstructure steel. The reinforced concrete structures affected consist of walls, columns, and floor slabs, which form the permanent building superstructure.

The temperature zones are established on the basis of the color of the structures of different materials of construction, melting of metals and burning of other materials. The extent of damage of each structure is evaluated on the basis of the temperature to which the structure was subjected and the physical condition of the structure.

The criteria for the requirement of replacement or repair of damaged

structure are as follows:

1. Steel Structures

Structural steel components will be replaced or will continue to be used depending upon whether their thermal sensitivity is lower or higher than the temperature to which the structural component was subjected during the cable fire. In addition any steel structure found to have been immersed in a temperature zone of 1000 F or more shall be replaced.

2. Concrete Structure

For reinforced concrete structures, the procedures for the evaluation of the concrete will consist of detailed visual inspection, comparison of temperature zones and testing of samples from core borings. For reinforcing steel, the procedures of evaluation will consist of the following:

(a) Analyze the adequacy of the structure by neglecting the steel bars exposed to high temperatures, and if found adequate no further evaluation will be done; and,

(b) Conduct sampling and testing of steel bars to determine their material characteristics. The measured material characteristics will be evaluated by rechecking the design of the affected structure. If the reduced characteristics are still adequate to resist the loads imposed on the structure, no modifications are necessary.

On the basis of the above criteria, the floor slab at elevation 621.25 in the reactor building which appeared to be most seriously damaged was evaluated by TVA. Reinforcing steel bar and concrete core samples were taken and tested. The results of the tests show that the material characteristics of the reinforcing steel and concrete were not adversely affected by the fire. On the basis of these findings, TVA concluded that the fire damage to reinforced concrete is structurally minor. Consequently, the repairs to be undertaken are considered non structural.

We have concluded that the criteria, methods and procedures that are used in the evaluation of damage of structures and in the replacement of structural steel components and repair of reinforced concrete structures will asssure that all structural components affected by the fire will be returned to their originally acceptable condition. Accordingly, since no facility modification is involved in the repair work, the work was approved by our letter dated August 28, 1975.

2. Approval of electrical design changes including circuit changes and permission to proceed with the restoration and installation of cable trays, conduit, and cables, including approval for splicing. The work and design changes are described in parts VI and X of the Restoration.

Basis for NRC Approval

The basis for the approval of this work is provided in Sections 3.0 and 4.0.

3. Approval for installation of the fire detection system and for the installation of distribution piping of the fixed spray system. These systems are described in subsection of Part X of the Recovery Plan.

Basis of NRC Approval

The basis for the approval of this work is provided in Sections 3.0 and 4.0. A number of additional modifications were requested regarding the installation of the spray system as proposed above. The modifications include features required to make the fixed spray system capable of automatic actuation. Additional automatic fixed spray systems may also be required. The approval and performance of the work identified above does not preclude these additional requirements. The approval of the final disposition of certain of the commitments made by TVA in their letter dated August 29, 1975 will be the subject of a later safety evaluation prior to the return to operation. Some of the commitments require that TVA investigate and make proposals regarding certain areas of system improvement; (i.e., additional fixed water spray coverage, fixed water spray or alternative in the the cable spreading room, modification of cable spreading room ventilation system, additional penetration seal material). The later safety evaluation, prior to authorization for return to operation will address the final resolutuion of these items. There also may arise during the course of the restoration work additional items that will require resolution and these will also be addressed at that time.

6.0 Conclusions

As discussed in our Safety Evaluation issued in connection with Amendment 10 to DPR-33 and Amendment 7 to DPR-52, June 13, 1975, the requirements of the Technical Specifications issued on that date provide for protection of the fuel, which has been placed in the storage pools. These requirements assure ample safety systems to protect the fuel from damage. Such systems were made independent of systems involved in the restoration work and eliminate the potential for adverse interactions resulting from removal of fire damaged systems and restoration of the facility.

The replacement and repair work authorized in connection with this safety evaluation, will provide for replacement of cabling equivalent to that provided in the original installation. The design changes and improvements in circuit separation and resulting improved fire protection, along with the design improvements in fire detection systems and fire extinguishing systems proposed by TVA, substantially enhance the capability of the facility to withstand fires and will provide reasonable assurance that the public health and safety will not be endangered. These changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin.

Based on the foregoing considerations, we have concluded that there is reasonable assurance that the public health and safety will not be endangered by the proposed changes and the restoration activities authorized herein and that such changes do not involve significant hazard considerations.

Our review of the safety of the facility to return to operation is still in progress and additional changes may be required as a result of our continuing review, or as a result of the additional studies which we have indicated are needed and which TVA has proposed as discussed above. This safety evaluation does not address resumption of operation at Browns Ferry Units 1 and 2. Any authorization to resume operation will be considered in a subsequent safety evaluation which will also consider necessary changes to plant operating technical specifications required in connection with such authorizations.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259 AND 50-260

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-33 and Amendment No. 11 to Facility Operating License No. DPR-52 issued to Tennessee Valley Authority for operation of the Browns Ferry Nuclear Plant, Units 1 and 2, located in Limestone County, Alabama. The amendments are effective as of their date of issuance.

The amendments modify the licenses to authorize modifications to Units 1 and 2 in conformance with "Plan for Evaluation, Repair, and Return to Service of Browns Ferry Units 1 and 2 (March 22, 1975 Fire)" in accordance with the licensee's request dated August 29, 1975. These amendments do not authorize return to operation of Units 1 and 2. That authorization will be the subject of another action upon completion of our review of the total restoration work required.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendments dated August 29, 1975, (2) Amendment No. 14 to License No. DPR-33 and Amendment No. 11 to License No. DPR-52, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 2nd day of September 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

ATTACHMENT V

FIRES AT U.S. AND FOREIGN NUCLEAR PLANTS

FIRES AT U.S. NUCLEAR PLANTS

Peach Bottom Unit 1	During construction activities on February 3, 1965, a cable fire was started as a result of sparks from a welding operation. One of the design changes instituted as a result of the fire was to replace cable using a polyethylene inner sheath with butyl-rubber sheathed cable. A second fire took place on April 13, 1967. The fire resulted from the ignition of polyurethane thermal insulation on piping associated with the low temperature delay beds and regenerative heat exchangers, components of the helium purification system.
San Onofre Unit 1	There were two fires in 1968; both involved cable trays. The fire on February 7, 1968, resulted from mechanical and electrical overloading of power cables for the pressurizer heaters. The fire on March 12, 1968, resulted from cable tray loadings in excess of design specifications.
Nine Mile Point Unit 1	During the startup testing in 1969 and 1970 a fire occurred which resulted in the destruction of motor control centers. Loose terminations on line conductors at the contactors and breakers caused the fire.

4.	Indian Point Unit 2	There was a construction fire involving wood scaffolding which burned the cable trays and was reported as a personnel error. The fire occurred on November 4, 1971.
5.	Quad Cities Unit 2	There was a fire in two electrical trays in the reactor building. The fire was confined to approximately 5 feet at the end of two cable tray sections. This event occurred July 16, 1972.
6.	Beaver Valley Unit 1	A fire occurred at a motor control center on October 4, 1972, apparently resulting from a short circuit in one of the splices in the temporary wiring for the heat-light bulbs in the base of the motor control center.
7.	Oconee Unit 1	A fire took place August 20, 1970, during construction activity near the reactor coolant pump suction piping. Oil from machining operations ignited and burned, killing one man and injuring three others. A second fire took place in the control rod drive transfer cabinet on March 9, 1972. The fire was caused by a loose electrical connection. Damage was limited to a few components within the cabinet.

A third and fourth took place on December 30, 1972. These fires were caused by the ignition of oil spilled from the reactor coolant-pump oil system when the oil came in contact with reactor coolant systems components.

8. Oconee Unit 2

A fire took place in the turbine building on June 27, 1975. Oil leaking from the turbine oil purifier system ignited when it came in contact with the purifier heaters. Cables above the fire were charred; however, there was no loss of operational circuits. A report is being prepared by the utility.

9. Salem Unit 1

A construction fire occurred in April, 1974, in a cable tray located in the 4 KV switchgear room of the auxiliary building. The fire occurred during a welding operation involving a piece of weld slag which fell onto a cable tray.

10. Salem Unit 2

A fire took place on October 18, 1974 during construction activities. Wood construction forms were ignited and burned cables in the area of the heavy equipment hatch of the containment building.

11. Browns Ferry Unit 1 & 2 — A fire occurred on March 22, 1975 in the cable spreading room during a penetration leak test; a candle ignited polyurethane foam material used to seal penetration cable openings.

FIRES AT FOREIGN NUCLEAR PLANTS

1. Muhleberg, Switzerland — During power ascension testing a turbine-generator oil fire occurred. This was a result of failure of a screwed pipe connection of the pressurized oil pipe leading to a control valve. The oil ignited and the fire spread to cables in the immediate area. This fire spread through unprotected cable tray openings into adjacent cable rooms. The fire occurred near the end of 1971.

2. Hinkley, England — Due to failure of a hydraulic relay valve of a turbine, oil sprayed out and was ignited by an adjacent hot steam valve. The fire at the turbine was extinguished quickly by means of a manually controlled water spray system as well as a hose line from an inside hydrant and portable

extinguishers. Burning oil flowing along cable trays caused serious damage to the cables. This fire took place in July, 1966.

3. Tokai Mura/Unit 1, Japan

Due to failure of a cock in a strainer assembly in the reactor building a major quantity of oil sprayed out when a strainer was being changed, and was ignited by a hot pipe handling 360° C steam. The sprinkler system put out the fire in four minutes. The reactor was not damaged but was shut down while repairs were made to the strainer assembly. The fire took place in 1967.

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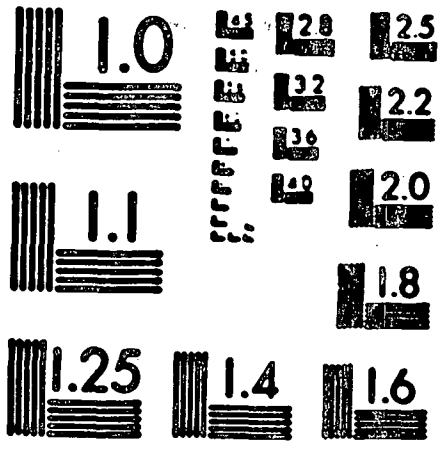

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