ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT
IEB 79-01B

TECHNICAL EVALUATION REPORT

DOCKET NO. 50-315

DATED: November 10, 1980

Licensee: Indiana and Michigan Electric Co.
Type Reactor: W PWR
Plant: D. C. Cook, Unit 1

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Introduction

This report is submitted in accordance with TI 2515/41[1/] for use as input to the Safety Evaluation Report on qualification of Class 1E electrical equipment installed in potentially "harsh" environmental areas at this facility.

Background and Discussion

IE Bulletin No. 79-01[2/] required the licensee to perform a detailed review of the environmental qualification of Class 1E equipment to ensure that the equipment would function under (i.e. during and following) postulated accident conditions.

The Technical Evaluation Report (TER) is based on IE's review of the licensee's submittal for conformance with the DOR guildelines or NUREG-0588, a site inspection of selected system components, to verify accuracy of the submittal, and EQB's review of component test reports.[3/]

Licensee submittals were received on March 7, 1980 and October 31, 1980.

The site inspection was completed on June 17, 1980.[4/] Generic and site specific guidance was requested from IE/NRR headquarters.[5/]

Summary of Licensee Actions/Statements

Aging is a generic industry issue whose resolution is not clear at this time. As relevant information becomes available, the licensee intends to continue his evaluation of the sensitivity to aging of materials and components.

Limit switches used only for indication of valve position will not be replaced. The limit switches, along with their cable and their cable terminations, for the Ice Condensor Refrigerant supply (VCR's) Main Feedwater Regulating Valve, and Steam Generator Stop Valves, have been deleted from the submittal. For the valves outside containment, valve position can be readily verified. Those inside containment (VCR's) are containment isolation valves; a redundant containment isolation valve located outside the containment, in series with the one inside the containment, will serve as a backup device. The position of the valve outside containment can be verified.

Equipment required due to changes in emergency operating procedures after January 14, 1980 (the issue date of IE Bulleitn 79-01B) is not always included in this submittal. The cut-off date was agreed to at the February 7, 1980 clarification meeting in Glen Ellyn, Illinois.

1/ Technical Evaluation Report (TER) On Results Of Staff Actions Taken To Verify Reactor Licensee Response To IEB 79-01B And Supplemental Information.
2/ Environmental Qualification of Class 1E Equipment.
3/ Attachment 1.
4/ Attachment 2.
5/ Attachements 3a and 3b.

In the equipment qualification charts, the specified radiation dose is the calculated bounding dose from gamma radiation as per the licensing basis of the Cook Plant. (Not the guidelines)

System Comparison

A comparison was made between the systems list provided by the licensee[6] and a similar list provided to IE by NRR[7] during a meeting in Bethesda, MD on September 30, 1980. The following systems were not included in the licensee's submittal.

- Auxiliary Steam Isolation
- Auxiliary Feedwater Isolation
- Containment Air Purification/Cleanup
- Accumulator
- Pressurizer Spray
- Power Operated Relief Valves
- Steam Dump
- Containment Radiation/Sampling
- Containment Sump
- Safety Equipment Area Ventilation

Equipment Evaluation

Class 1E equipment was evaluated, that is, placed into five separate categories.[8] Result of the evaluation follows: (See pages following)

Caveat

Test reports and other documentation which licensees referenced as establishing environmental qualification were reviewed for acceptability by NRR, Environmental Qualification Branch. (Reference Attachment 3a, memorandum dated June 20, 1980 Hayes to Jordan.)

This TER does not include information about seismic or fire withstand capability. It should therefore not be inferred that Category I equipment meets all necessary qualification requirements.

Conclusion

Based on IE's review of the licensee's submittal, the site inspection, and licensee's proposed actions, it cannot be concluded that there is reasonable

6/ Attachment 4.
7/ Attachment 5.
8/ Attachment 6.

assurance all components installed at the Donald C. Cook Nuclear Plant, Unit 1 are environmentally qualified and installation methods of environmentally qualified components would not contribute to the failure of such components during a potential accident.

A positive conclusion cannot be made until:

1. All matters referred to IEHQS/NRR have been satisfied.9/

2. The 10 systems missing from the licensee's submittal have been evaluated by NRR: (Page 2)

3. The negative equipment evaluations have been reviewed by NRR. (Pages 4, 5, 6, and 8.)

9/ Attachment 8.

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS:	RH	SPRAY	RAD	AGING	ATTI REF#	CONCUR [10]
Ia 1	Cont. Cable	Continental	XLPE/ASB 3119	Below Flood	AUX/ CNT	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,F,I,K, P,Q,Y
Ia 2	Cont. Cable	Continental	XLPE/ASB 3120	—	AUX.	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,L R
Ia 3	Cont. Cable	GE	XLPE/ASB 3120	—	AUX	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,L, X
Ia 4	Power Cable	Anaconda	3120	—	AUX	Yr	340	119.7	100	2000PPM	$2X10^8$	40yrs	5	Yes-A
Ia 5	Cont. Cable	Continental	XLPE/ASB 3121	Flood up Tubes	AUX/ CNT	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,K, P,Q.
Ia 7	Cont. Cable	GE	XLPE/ASB 3121	Flood up Tubes	AUX/ CNT	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,K, P,QX
Ia 8	Cont. Cable	Continental	XLPE/ASB 3122	Below Flood	AUX/ CNT	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,K, P,Q,Y
Ia 9	Cont. Cable	GE	XLPE/ASB 3122	Below Flood	AUX/ CNT	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,Y,X, I,K,P,Q
Ia 10	Cont. Cable	Continental	XLPE/ASB 3123	Below Flood	AUX	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-R,I,K, P,Q
Ia 11	Cont. Cable	GE	XLPE/ASB 3123	—	AUX	1Yr	345	121.7	100	2500PPM	$1.5X10^8$	40yrs	8	NO-C,I,L, X
Ia 12	Power Cable	Kerite	3127	—	CNT	Yr	325	96.7	100	2600PPM	$2X10^8$	—	7	NO-H,I,K
Ia 13	Inst. Cable	BIW	XLPE/CSPE 3064	Below Flood	AUX/ CNT	1Yr	290	59.7	100	2000PPM	$2X10^8$	250°/7day	31 & 32	NO-Y,N,I, L
Ia 14	Inst. Cable	Rockbestos	3064	Below Flood	AUX/ CNT	1Yr	346	127.7	100	3000PPM	$2X10^8$	—	34	NO-Y,M,I, K
Ia 15	Inst Cable	Samuel Moore	3075	Below Flood	AUX/ CNT	1Yr	340	119.7	100	3000PPM	$2X10^8$	250°/7day	10 & 11	NO-Y,N,I, K
Ia 16	Inst. Cable	Continental	3075 XLPE/Hypalon	—	AUX	1day	340	114.7	100	—	$1X10^7$	40yrs	33.	Yes-A

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT REF	CONCUR [19]
Ia 17	Inst. Cable	BIW	Hyp/Hyp 3075	Below Flood	AUX/CNT	1yr	345	121.7	100.	2500PPM	1.5×10^8	40yrs	8	NO-Y, I, K
Ia 18	Inst. Cable	Cerro	3077	Below Flood	AUX/CNT	1yr	346	127.7	100.	3000 PPM	2×10^8	300°/54days	12	NO-Y, I, K, N
Ia 19	Inst. Cable	Samuel Moore	3077	Below Flood	AUX/CNT	1day	340	119.7	100	3000PPM	2×10^8	250°/7day	10 & 11	NO-Y, I, K, N
Ia 20	Inst. Cable	Continental	3077	—	AUX	1day	340	114.7	100	—	1×10^7	40yrs	33	yes-A
Ia 21&22	Power Cable	OKonite	Okonite/okoprene 324	—	AUX/CNT	1yr	346	127.7	100.	3000PPM	2×10^8	IEEE 383-1974 2.3.3.2	49	yes-A
Ia 24	Power Cable	OKonite	399	Flood up Tubes	AUX/CNT	1yr/:	346	127.7	100	2000PPM	2×10^8	240°/7day	6	NO-I, K, N
Ia 26	Inst. Cable	OKonite	EPR 3102	—	AUX	1yr/:	346	127.7	100.	IEEE 323-74	2×10^8	IEEE 383-74	37	No-I, L, P, Q
Ia 27	Power Cable	Anaconda	3116	Flood up Tubes	AUX/CNT	1yr/:	340	119.7	100.	3000PPM	2×10^8	250°/7day	5	NO-I, K, N
Ia 28	Power Cable	Essex	3116	Flood up Tubes	AUX/CNT	1yr/:	346	127.7	100	3000PPM	2×10^8	250°/7day	25	NO-I, K, N
Ia 29	Power Cable	Kerite	3116	Flood up Tubes	AUX/CNT	6MO/:	325	96.7	100	2600PPM	1.2×10^8	—	7	NO-I, K, M
Ia 31	Penetration	Conax	EPI 4KV Power	Below Flood	CNT	1yr/3days	346	122	100	2098 PPM	1×10^8	None	3, 17, 18	NO-Y, I, M
Ia 32	Penetration	Conax	EP-2 thru EP-14	Below Flood	CNT	1yr/6hrs	340	116	100	2098 PPM	1×10^8	None	1, 3, 17, 18	NO-Y, I, M
Ia 33	Fan Motor	W	TBDP	—	CNT	1yr/none	320	89.7	100	2500PPM	2×10^8	390°/20days	21	No-C
Ia 45	H_2 Recombiner	W	—	—	CNT	3MO/none	310	77	100	2500PPM	2×10^8	80 start ups	20	No-M
Ia 48	Main FW Flow XMTR	Foxboro	E13DM-HSAH1 (MCA)	—	AUX	1.5 & 25sec/≤1.5	290	75	100	—	—	—	WCAP 8541	yes-A

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS.	RH	SPRAY	RAD	AGING	ATT 1 REF#	10/ CONCUR
Ia 63	Przr. Press. XMTR	Barton	763 *	List E 10/	CNT	≤ 3 sec	360	90	100	H_3BO_3-NaOH	$5X10^7$	—	30	No-M
Ia 57	Main Stm Flo XMTR	Barton	764	(-) Flood 10/ List A	CNT	≤ 3 Sec	360*	90	100	H_3BO_3-NaOH	$5X10^7$	—	30	No-M
Ia 59	Main Stm Press XMTR	Foxboro	E11GM-HSAE1 (MCA)	10/ List B	AUX	≤3 sec	318	90	100	—	$5X10^7$	—	NS-PLC-5023	No-M, Z
Ia 62	Press. Level XMTR	Barton	764	10/ List J	CNT	4 MOS.	160*	90	100	H_3BO_3-NaOH	$5X10^7$	—	30	No-M
Ia 64	Przr Press XMTR	Barton	763 *	10/ List E	CNT	≤ 3 Sec	320	90	100	H_3BO_3-NaOH	$5X10^7$	—	30	No-M
Ia 65	Przr Press XMTR	Barton	763 *	10/ List E	CNT	4 MO	160*	90	100	H_3BO_3 NaOH	$5X10^7$	—	30	No-M
Ia 66	Rx Clnt Press XMTR	Barton	763 *	(-) Flood Post Acc. Monit.	CNT	4 MO	330*	90	100	H_3BO_3 NaOH	$7.6X10^5$	—	30	No-M
Ia 67	Rx Clnt Press XMTR	Barton	763 *	(-) Flood Post Acc. Monit.	CNT	4 MO	280*	90	100	H_3BO_3 NaOH	$5X10^7$	—	30	NO-M
Ia 68	Przr Press XMTR	Barton	763 *	Long Term Post Acc	CNT	4 MO	160*	90	100	H_3BO_3 NaOH	$5X10^7$	—	30	NO-M
Ia 69	Rx Clnt Temp XMTR (RTD)	Sostman/Rosemont	11834B/176KF	10/ List D	CNT	≤ 5/15"	320*	90	100	H_3BO_3 NaOH	$1X10^8$	—	28	No-M
Ia 70	Rx Clnt Temp XMTR (RTD)	Sostman/Rosemont	11834B/176KF	10/ List D	CNT	≤ 5"	330*	90	100	H_3BO_3 NaOH	$1X10^8$	—	28	NO-M
Ia 72	Rx Clnt Temp XMTR (RTD)	Sostman/Rosemont	11901B/176KS	Post Acc. Monit.	CNT	4 MO	330	90	100	H_3BO_3 NaOH	$1X10^8$	—	28	No-M
Ia 78	Motors - CVCS, SI-RHR	W	5808Z, 5009H, 500.9-P24	—	AUX	—	—	—	—	—	$2X10^8$	390°/20 day	21	No-N,C
Ia 79	Motor - CS	Reliance	Frame 5810F	—	AUX	—	—	—	—	—	$1X10^8$	—	None	No-M
Ia 86	Stm. Gen. Stop Vv. Dump	ASCO	HT8316B17	SOL. VV.	AUX	5 sec	180	14.7	0	—	—	—	ASCO Cat. No. 30	No-M

10/ Attachment 9 *The same component is "qualified" for several temps.

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATTI REF#	CONCUR
Ia 96	Cntmt Iso	ASCO	NP-8316S4V	Sol. Vv.	CNT	≤ 10 sec	—	—	100	H_3BO_3 NaOH	1.5×10^8	4	29	No-Q
Ia 97	Control Cable Term.	—	strnd. Kapton to XLP	—	CNT	—	345	125	100	2500ppm	1.5×10^8	302°/4day	8 & 14	No-N,I
Ia 98	Cable Term	—	XLP-XLP	—	CNT	—	345	125	100	2500ppm	1.5×10^8	302°/7day	8 & 14	No-N,I
Ia 99	Cont. Cable Term	—	—	Mot. Op. Vvs.	AUX	—	212	14.7	100	—	—	—	44	No-N,I
Ia 100	Cont. Cable Term	—	—	—	AUX	—	340	119.7	100	2600ppm	2.04×10^8	360°/4days	22 & 23	No-R,N,I
Ia 101	Cont. Cable Term	—	—	Mot. Op. Vvs.	AUX	—	250	39.7	100	—	2.04×10^8	165°/8days	24	No-R,N,I
Ia 103	Term @ T.B.	—	—	—	AUX	—	345	124.7	100	2500ppm	2×10^7	—	45 & 46	No-R,M,I
Ia 104	Term	—	—	—	CNT	—	330	104.7	100	2600ppm	1×10^7	360°/4days	1 & 22	No-N,I
Ia 106	Term @ Sol. Vvs.	Deleted	—	—	CNT	≤ 10 sec	—	—	100	H_3BO_3 NaOH	1.5×10^8	4	29	—
Ia 107	Term @ Barton Inst.	—	—	—	CNT	—	340	119.7	100	2000ppm	1.5×10^8	250°/7days	8,9,10,11,12,14,17,18,19	No-I,N,W
Ia 108	Term @ RTD	—	—	—	CNT	—	340	119.7	100	2000ppm	1.5×10^8	250°/7days	8,9,10,11,12,14,17,18,19	No-I,N,W
Ia 109	Penet. Term	—	—	—	CNT	—	340	118	100	2500ppm	1.5×10^8	—	13	No-I,M
Ia 110	Inst. Cable Term @ Penet	—	—	—	CNT	—	340	119.7	100	2000ppm	1.5×10^8	250°/7day	8,9,10,11,12,14,18,19,17	No-I,N
Ia 113	Power Cable Term	—	Solid Kapton to Strnd Kapton	—	CNT	—	340	118	100	2500ppm	6×10^7	—	13	No-M,I
Ia 114	Term	—	Strnd Kapton to Strnd Hyp.	—	CNT	—	340	118	100	2500ppm	1.5×10^8	—	13	No-MI

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATTI REF[1]	CONCUR[10]
Ia 115	Term-MOV, H_2R, Fan Mot	—	—	—	AUX CNT	—	340	118	100	2500PPM	$1.5X10^8$	—	13	No-Y, I, M
Ia 116	Term @ Mot	—	—	—	AUX	—	—	—	—	—	$1.5X10^8$	—	13	No-R, I, M
Ia 117	MOV	Limitorque	SMB-1, 00, 2	ECCS INJ./ RHR Clg	CNT	6 days	340	119.7	100	2600PPM	$2.04X10^8$	360°/4days	22 & 23	No-Z, N
Ia 118	MOV	Limitorque	SMB-1	SI	CNT	30 days	315	84.7	100	3000PPM	$2.64X10^8$	360°/4days	16	No-Z, N
Ia 119	MOV	Limitorque	SMB-000	Air Recirc Dampers	CNT	16 days	250	39.7	100	—	$2.24X10^8$	165°/8 days	24	No-Z, N
Ia 120	MOV	Limitorque	SMB-00	RCP Seal Water	CNT	6 days	330	104.7	100	2600PPM	—	360°/4days	22	No-Y, N, Z
Ia 121	MOV	Limitorque	—	—	AUX	16 days	250	39.7	100	—	$2.24X10^8$	165°/8days	24	No-N, Z
Ia 122	MOV	Limitorque	—	—	AUX	6 days	340	119.7	100	2600PPM	$2.04X10^8$	360°/4days	22 & 23	No-N, Z
Ia 63	Prgr press XMtr.	Barton	763	Post Acc. Monit. IV List E	CNT	≤ 3 sec	360	89.7	100	H_3BO_3 NaOH	$5X10^7$	—	30	No-I, M
Ib 23	Power Cable	Essex	EPR/Neop. 324	—	AUX	1 day	194	—	—	—	$1X10^6$	—	Eng. Rev	No-P, M
Ib 25	Power Cable	Anaconda	EPR/HYP 3102	—	AUX	1 day	194	—	—	—	$1X10^6$	—	Eng Rev	No-P, M
Ib 30	Power Cable	Anaconda	3103	—	AUX	1 year	194	—	—	—	$1X10^6$	—	Eng Rev	No.
Ib 34	Grease	Mobil	Mobilux EP-2	—	AUX CNT	1 year	250	—	100	—	$2.4X10^7$	—	47 + 48	Yes.
Ib 35	Grease	Mobil	Mobilux #2	—	AUX	—	250	—	—	—	$1X10^8$	—	48	Yes.
Ib 36	Oil	Mobil	DTE OIL Medium	—	AUX	—	250	—	—	—	$1X10^8$	—	48	Yes

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT'S REQ'D	CONCUR
I,b 37	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 38	Oil	Mobil	DTE 797O.1	—	AUX	—	410	—	—	—	1×10^8	—	48	Yes
I,b 39	Grease	Mobil	—	—	AUX	—	250	—	—	—	1×10^8	—	48	Yes
I,b 40	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 41	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 42	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 43	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 44	Grease	Mobil	Sovarex L:O	—	AUX	—	400	—	—	—	1×10^8	—	48	Yes
I,b 50	Mini Flow Controlr	Barton	289A/199	RHR PUMP	AUX	Cont	200	14.7	wthr Prf	—	—	—	Eng. Rev.	Yes-S
I,b 54	Pump Disch Flow	Barton	332	SI	AUX	Cont	160	14.7	Exp Prf	—	—	—	Eng. Rev	Yes-S
I,b 55	HTX. Flow	Foxboro	E13DM-HSAW	RHR	AUX	Cont	180	14.7	NEMA 4	—	—	—	Eng Rev	Yes-S
I,b 56	—	—	—	—	—	—	—	—	—	—	—	—	—	—
I,b 105	Cable Term	—	—	Sol. Vvs.	AUX	5 sec	180	14.7	—	—	—	—	Eng. Rev.	V-B
IIa. II.b 102	None Cable Term			Prgr, Rlf. Vvs	CNT	1yr	250	39.7	100	—	2.14×10^8	165°/8days	24	Yes-D
III 46	Stm Gen Lev Xmtr	Barton	764	[9] List H	CNT	4MO	160	89.7	100	H_3BO_3 $NaOH$	5×10^7	—	30	B-T

CAT		DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT 1 REF#	[10] CONCUR.
III	47	Cond. Stor Tk Lev. Xmtr	Taylor	304TD00212	—	Aux	—	—	—	—	—	—	—	—	B-E
III	51	BIT Flow XMtr	Foxboro	E13DH-HSAH1 MCA	Monitor (-) Flood	CNT	90 min	290	74.7	100	H_3BO_3 NaOH	7.6×10^7	—	AEW 8541	B-T
III	52	BIT Flow Xmtr	Foxboro	E13DH-HSAH1	Long Term Monit. (-) Flood	CNT	90 min	290	74.7	100	H_3BO_3 NaOH	7.6×10^7	—	AEW 8541	B-T, O
III	53	—	—	—	—	—	—	—	—	—	—	—	—	—	—
III	60	Int. Rng Flux	—	—	—	—	—	—	—	—	—	—	—	—	Yes
III	61	Power Rng Flux	—	—	—	—	—	—	—	—	—	—	—	—	Yes
III	73	Cntmt Press	Foxboro	E11GM-HSAA1	[9] List F	Aux	—	—	—	—	—	—	—	—	Yes-S
III	74	Cntmt Press	Foxboro	E11GM-HSAA1	[10] List G	Aux	—	—	—	—	—	—	—	—	Yes-S
III	75	Radiation Monit.	W	1101	—	CNT	—	120	28.7	100	—	—	—	—	Yes-S
III	90	Air Operator Diaphrm	Fisher	667	CNT FW ISOL	Aux	—	—	—	—	—	—	—	—	Yes-E
III	81	Transducer	Fisher	546	Feed water	Aux	320	90	100	—	—	—	—	—	Yes-T
III	83	Air Operator Diaphram	Bailey	656	RHR HTX	Aux	—	—	—	—	—	—	—	—	Yes-E
III	84	Steam Piston	Atwood-Morell	—	Main Stm ISO	Aux	—	—	—	—	—	—	—	—	Yes-E
III	85	Air Operator Diaphram	Fisher	657	stm Gen St Vv Dump	Aux	—	—	—	—	—	—	—	—	Yes-E
III	87	Air Operator Diaphram	Fisher	667	Mn. Stm Pr. Rlf.	Aux	—	—	—	—	—	—	—	—	Yes-E

[10] Attachment 9

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATTI REF[1]	CONCLR[1]
III .88	Transducer	Fisher	546	Mn. Stm Press Rlf.	AUX	—	320	90	100	—	—	—	—	Yes-T
III .89	Air Operator Diaphram	Masoneilan	37 & 38	Przr PORV	CNT	—	—	—	—	—	—	—	—	Yes-E
III 90	Sol Cont. Vv	ASCO	NP-8316S4V	Przr Press	CNT	—	346	110	100	H_3BO_3 $NaOH$	1.5×10^8	4	30	No-Q
III .91	Air Operator Diaphram	Grinnell	3250	Ice Cond. Ref.	CNT	—	—	—	—	—	—	—	—	Yes-E
III .92	Air Operator Diaphram	Bailey	656	Inst. Rm Purg/Sup	CNT	—	—	—	—	—	—	—	—	Yes-E
III 93&94	Air Operator Diaphram	Bailey	722 & 732	Cntmt Purg/Exh	CNT	—	—	—	—	—	—	—	—	Yes-E
III .94	—	—	—	—	—	—	—	—	—	—	—	—	—	—
III 95	Air Operator Diaphram	Bailey	656	Cntmt Press Rlf	CNT	—	—	—	—	—	—	—	—	Yes-E
III 111	Termination Inst	—	—	Foxboro	CNT	—	320	89.7	100	2000ppm	1.8×10^8	—	26 & 27	Yes-T
III 112	Termination Inst	—	—	Mercoid	Aux	—	—	—	—	—	—	—	—	Yes-S
IV a	None	—	—	—	—	—	—	—	—	—	—	—	—	—
IV b 123	MOV	Limitorque	—	—	Aux	28 hr	212	14.7	100	—	?	—	44	No-Z
IV b 125	MOV	Limitorque	SMB-2	RHR Suct	CNT	6 days	330	104.7	100	2600ppm	?	360°/4days	22	No-Z
V 49	Aux FW Flow Xmtr	Barton	368	To be replaced	Aux	—	—	—	—	—	—	—	—	Yes-D
V 76	Ess. Serv. Wat. P.P. St.	Mercoid	DA-7031-153	To be replaced	Aux	—	—	—	—	—	—	—	—	Yes-D

[1] Attachment 9

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT1 REF#	CONC[1]
Ia 77	Valve Posit Ind.	NAMCO	EA180	To be installed PZR PORVs	CNT	30 days	340	84.7	100	3000 PPM	2.04×10^8	200°/8 days	43	Yes-D
II 82	Reg. Vv. Trip	ASCo	NP-8300C5BRU HT-8300B5BRU	SOL. VV. MFW	AUX	5 sec	160	—	—	—	—	—	ASCO Cat 30	Yes
II 124	MOV	Limitorque	SMB-00	PZR PORV	CNT	16 days	250	39.7	100	—	2.24×10^8	165°/8 days	24	NO-Z,N

LIST OF TEST REPORTS

1. IPS-234 Penetration
2. Blank
3. IPS-137 Penetration
4. Blank
5. F-C 3341 Anaconda Cable
6. F-C 3694 Okonite Power Cable
7. Kerite Report on Effects of Gamma Rad and Autoclaving Power Cable
8. Conax IPS-348 BIW Cable
9. F-C4033-1 Terminations
10. F-C3683 Sam Moore Cable
11. Isomedix 5/76 Sam Moore Cable
12. Cerro Cable 5/76
13. CWAPD-332 Terminations
14. F-C 4033-3 Term - Kapton to XLP
15. Blank
16. MOV 600456
17. IPS-326
18. IPS-327
19. IPS-329 Term
20. WCAP 7709-L Supp 2 H_2 Recombiner
21. WCAP 7829 Fan Motors TBDP
22. MOV 600198
23. MOV 600376A
24. Limitorque 600461
25. Isomedix Corp. T.R. 11/75 Essex Power Cable
26. Foxboro T.R. No. TE-1013
27. Letter W to Case 4/26/78 No. NS-PLC-5023
28. WCAP 9157 RTDs
29. ASCO AQS 21678
30. Barton XMTRS 764 W Letter NS-TMA-1950
31. BIW 73C212 Cable
32. BIW 75C008 Cable
33. F-C 2935 Continental Cable
34. Rockbestos Firewall III
35. FC3016 Cyprus Power Cable
36. Blank
37. Qual. of Okoguard EPR Insul. (Okonite) Inst. Cable
38. Cyprus Statement of 6/16/76
39. Cyprus Statement of 9/14/76
40. Cyprus Report #3525
41. Cyprus Report #3658
42. Cyprus Statement of 6/21/76
43. NAMCO Limit SNS 9/5/78
44. F-C 3271 Control Cable MOVs
45. IPS-339 Conax Cable Term
46. IPS-349 Conax Cable Term
47. Grease Mobilux EP-2 Letter
48. Grease Mobilux Letter
49. Okonite Form N-1 Power Cable

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

July 1, 1980

MEMORANDUM FOR: E. L. Jordan, Assistant Director, Division of Reactor Operations Inspection, IE:HQ

THRU: G. Fiorelli, Chief, Reactor Construction and Engineering Support Branch

FROM: D. W. Hayes, Chief, Engineering Support Section 1

SUBJECT: SCREENING REVIEW OF LICENSEE RESPONSE TO IEB 79-01B AND SUMMARY OF INSPECTION OF INSTALLED SYSTEMS AT D.C. COOK UNITS 1 AND 2 - DOCKET NOS. 50-315; 50-316

Frank Jablonski has completed his initial review of the D.C. Cook facility response to IEB 79-01B, and the inspection phase of the system audit. There were some basic differences between Units 1 and 2, e.g., main steam flow and narrow range RCS temperature transmitters were not part of Unit 2's engineered safeguards actuation; Unit 1 did not have a "dedicated" post accident monitoring system. A telephone conversation with the licensee indicated the matters were considered by NRR during licensing of Unit 2.

The licensee has taken exception to aging, replacement of certain limit switches (indication only), and to protection of certain components from adverse environments including flood[1].

Test reports referenced by the licensee are included in Attachment 2. The reports were not evaluated as part of this review. Review of all test reports and other documentation referenced by the licensee shall be accomplished by NRR (Environmental Qualification Branch).

A walkdown was conducted on June 16 and 17, 1980 to inspect installed components associated with Unit 1 Engineered Safety Features actuation and Containment Isolation, and Unit 2 Containment Spray Systems. Prior to the walkdown, wiring diagrams were reviewed. Components included on Attachment 3 were observed.

[1] See Attachment 1
(Excerpts from Memo Hayes to Jordon A/I F03067180 dated June 20, 1980)

Onsite Inspection
ATTACHMENT 2

Unit 1 Observations:

Isolation Valves

Four solenoid valves were observed; all were ASCO catalog number 2063812RVU or NP831654V. (The control cables and penetrations for the solenoids inside containment were not located above flood level nor were cables installed in flood-up tubes). Limit switches, associated with valves controlled by the solenoid valves, were NAMCO number EA740-6000-0.

Two limitorque motor operated valves were observed. One was type SMB-00 with a Reliance motor, insulation Class RH, the other was type SMB-2 with a Reliance motor, no insulation class stamp. Both valves were located below flood level. One valve had its cable installed in flood-up tubes, the other did not.

Instruments

Four transmitters were observed, including flow and pressure. Temperature measuring elements could not be observed because of high radiation. With the exception of the containment pressure transmitter (outside containment) all Barton and Foxboro instruments included a hermetic seal. Only the cables for the Pressurizer Pressure (long term monitoring) transmitter were installed in flood-up tubes. The main steam pressure transmitters (outside containment) were protected by missile shields.

Terminations (See Attachment 4)

All terminations at instruments inside containment were covered with Raychem WCSF heat shrink tubing.

Solenoid terminations for valves VCR-103 and VCR-104 were covered by electrical tape in a junction box with gasketed cover. The terminations had just previously been made and were supposedly in a manner qualified in ASCO test report AQS 21978/TR.

Terminations at penetrations were various. Those penetrations which had cables in flood-up tubes were terminated in large terminal boxes at elevations above the maximum flood level. Terminating points were covered by Raychem WCSF heat shrink tubing.

For penetrations without cable in flood-up tubes, (non safety-related) control cable terminations were made to terminal blocks in large terminal boxes at elevations below maximum flood level. Instrument cable was terminated to the penetration pigtails by multi pin connectors or directly

with butt splices covered by Raychem WCSF tubing. Power cable (both safety and non safety-related) was terminated directly to the penetration pigtails with the termination point covered by Raychem heat shrink tubing.

Power, control, and instrument penetration pigtail insulation was spirally wound Kapton, and in many cases, was exposed directly to the potential containment accident environment of radiation, chemical spray, and flood. The concern is, even though a component may not be safety-related or required to operate subsequent to flooding, could short circuits cause gross failure of a penetration and breach containment integrity? (Failure of Kapton insulation is discussed in licensee reference No. 13; refer to I&M letter to Case dated April 21, 1978, last page).

Cables (See Attachment 4)

(Refer to discussion on terminations).

The only cable which had visible markings was BIW's Bostrad 7. Cables, like terminations, were installed in various ways including below flood level. Since none of the cable was tested for long term submergence, the same concern exists for cable as does for terminations, i.e., short circuit effects on penetrations. Most of the power and control cable had an asbestos braid jacket.

NOTE: Even though beyond the scope of IEB 79-01B, it is unclear whether in Unit 1 justification has been made for use of cables which did not pass the IPCEA-19-81 vertical flame resistance test. Reference: November 25, 1977 NRR letter to licensee, subject: "Summary of Meetings Held on November 3 and 4, 1977, Reference - Fire Protection" (Question 16).

Penetrations (See Attachment 4)

Penetrations were located below flood level. Penetrations were manufactured by Conax and consisted of power, control, and instrument types. Data sheets indicated that test operating times varied between 6 hours for instrument/control type, and 3 days for power type. Penetrations are specified to remain operable, i.e., integrity, for up to one year.

As reported above, penetration feed-throughs (pigtails) were terminated to plant cables in three distinct ways: multi-pin connectors; butt spliced directly and covered; through flood-up tubes via a separate terminal cabinet. Except for those pigtails in flood-up tubes, the Kapton insulated penetration feed-through conductors (pigtails) were exposed directly to potential harsh environments which could cause short circuits.

Fan Motor

A Westinghouse motor was coupled to fan unit HV-CEQ-2. The following pertinent nameplate data was obtained: Serial No. 7206, HP-75, Insulation F, Frame 449TZ, Type TBDP, Style 71C20763-04, Bearings 4109D41G05, Grease 53701RW - Style 773A773G05.

Page F1-1 of the licensee's June 5, 1980 submittal references Westinghouse Corp. test report WCAP-7829 as the qualification document. A cursory review of the report indicates it does not identify the specific motor type which was tested; however, the report specifically states that Thermalastic Epoxy insulation is a "key component" in the motor, yet the motor nameplate only identifies insulation Class F. Further, page F1-1 specifies the operating time to be one year; however, the "qualification" column of F1-1 is blank. The WCAP-7829 test was for "short term" which was not defined although inferred to be less than one year.

Unit 2

Components for the containment spray system were all located outside containment. Redundant components were located in separate rooms. A list of equipment observed and model numbers is included on Attachment 3. No adverse environments were apparent.

Conclusion

Except as noted above, cables - fan motor type - exceptions to aging - replacement of limit switches - and protection from adverse environment, equipment desciptions provided by the licensee on the system component evaluation work sheets were complete and accurate. As stated previously, all test reports listed by the licensee will be evaluated by the Environmental Qualification Branch of NRR.

D. W. Hayes, Chief
Engineering Support Section 1

Attachments:
1. Excerpts from Memo Hayes to Jordan (A/I F03067180) dtd 6/20/80
2. List of Test Reports
3. Components Inspected
4. Figures 1A and 1B

cc w/attachments:
J. G. Keppler
G. Fiorelli
R. Heishman
R. Masse, Res. Insp.
V. Thomas, IE:HQ

ATTACHMENT 2

ATTACHMENT 1

EXCERPTS FROM MEMO HAYES TO JORDON (A/I F03067180)
DATED JUNE 20, 1980

Item 3, Attachment 1 to Memo

Reference No. 3, even though a component may not be required to operate subsequent to flooding, what effect will short circuits have on containment electrical penetrations? Was this considered by NRR?

Items 2, 4, 5, Attachment 2B to Memo

2. Licensees maintain that aging is a generic industry issue whose resolution is not clear; therefore, evaluation has not been made or will continue to be made as relevant information is made available.

4. Limit switches used for valve position indication only have been deleted from the submittal. Licensees maintain that a valve outside containment in series with one inside can have its position verified visually following an accident. Is this acceptable?

5. The licensees maintain that neither valve position limit switches, solenoid valves, nor control cables for air operated containment isolation valves need be replaced or protected from the adverse environment, including flood, because all postulated failures will result in the isolation valve assuming its fail-safe position. Is this acceptable?

ATTACHMENT 2

LIST OF TEST REPORTS

1. IPS-234 Penetration
2. Blank
3. IPS-137 Penetration
4. Blank
5. F-C 3341 Anaconda Cable
6. F-C 3694 Okonite Power Cable
7. Kerite Report on Effects of Gamma Rad and Autoclaving Power Cable
8. Conax IPS-348 BIW Cable
9. F-C4033-1 Terminations
10. F-C3683 Sam Moore Cable
11. Isomedix 5/76 Sam Moore Cable
12. Cerro Cable 5/76
13. CWAPD-332 Terminations
14. F-C 4033-3 Term - Kapton to XLP
15. Blank
16. MOV 600456
17. IPS-326
18. IPS-327 (527)
19. IPS-329 Term
20. WCAP 7709-L Supp 2 H_2 Recombiner
21. WCAP 7829 Fan Motors TBDP
22. MOV 600198
23. MOV 600376A
24. Limitorque 600461
25. Isomedix Corp. T.R. 11/75 Essex Power Cable
26. Foxboro T.R. No. TE-1013
27. Letter W to Case 4/26/78 No. NS-PLC-5023
28. WCAP 9157 RTDs
29. ASCO AQS 21678
30. Barton XMTRS 764 W Letter NS-TMA-1950
31. BIW 73C212 Cable
32. BIW 75C008 Cable
33. F-C 2935 Continental Cable
34. Rockbestos Firewall III
35. FC3016 Cyprus Power Cable
36. Blank
37. Qual. of Okoguard EPR Insul. (Okonite) Inst. Cable
38. Cyprus Statement of 6/16/76
39. Cyprus Statement of 9/14/76
40. Cyprus Report #3525
41. Cyprus Report #3658
42. Cyprus Statement of 6/21/76
43. NAMCO Limit SNS 9/5/78
44. F-C 3271 Control Cable MOVs
45. IPS-339 Conax Cable Term
46. IPS-349 Conax Cable Term
47. Grease Mobilux EP-2 Letter
48. Grease Mobilux Letter
49. Okonite Form N-1 Power Cable

ATTACHMENT 3

COMPONENTS INSPECTED

D.C. COOK — UNITS 1 AND 2

SYSTEM	UNIT	PLANT ID NO.	GENERIC NAME	IN	OUT
ESS[1]	1	MFC-110[9]	Transmitter	X	
ESS[1]	1	MPP-220[10]	Transmitter		X
ESS[1]	1	NTP-111	Transmitter	X	
ESS[1]	1	NPP-151[11]	Transmitter	X	
ESS[1]	1	PPP-300[10]	Transmitter		X
CI[2]	1	QCM-250[7]	Motorized Valve	X	
CI[2]	1	ICM-111[8]	Motorized Valve	X	
CI[2]	1	FRV-210[5]	Solenoid Valve		X
CI[2]	1	VCR-101[6]	Solenoid Valve	X	
CI[2]	1	VCR-103[6]	Solenoid Valve	X	
CI[2]	1	VCR-104[6]	Solenoid Valve	X	
CS[3]	2	PP-009[14]	Pump/Motor		X
CS[3]	2	IMO-215[12]	Motorized Valve		X
CS[3]	2	IMO-210[12]	Motorized Valve		X
CS[3]	2	IMO-222[13]	Motorized Valve		X
CS[3]	2	ILS-950[15]	Transmitter		X
--	1	Various	Penetration	X	
--	1	Various	Cable	X	
--[4]	1	Various	Terminations	X	
AR[4]	1	HV-CEQ-2	Fan Motor	X	

[1] Engineered Safety Features

[2] Containment Isolation

[3] Containment Spray

[4] Air-recirculation

[5] Catalog No. 2063812RVU

[6] Catalog No. NP831654V

[7] SMB, Class RH, no flood-up tubes

[8] SMB-2, Class , flood-up tubes

[9] ITT Barton 764

[10] Foxboro E11GM

[11] ITT Barton 763

[12] Limitorque, SMB-0. Reliance Motor, Ins. Class HR

[13] Limitorque, SMB-00, Ins. Class B

[14] Reliance, Type P, Ins. Class B

[15] Foxboro E13DM without hermetic seal

ATTACHMENT 4

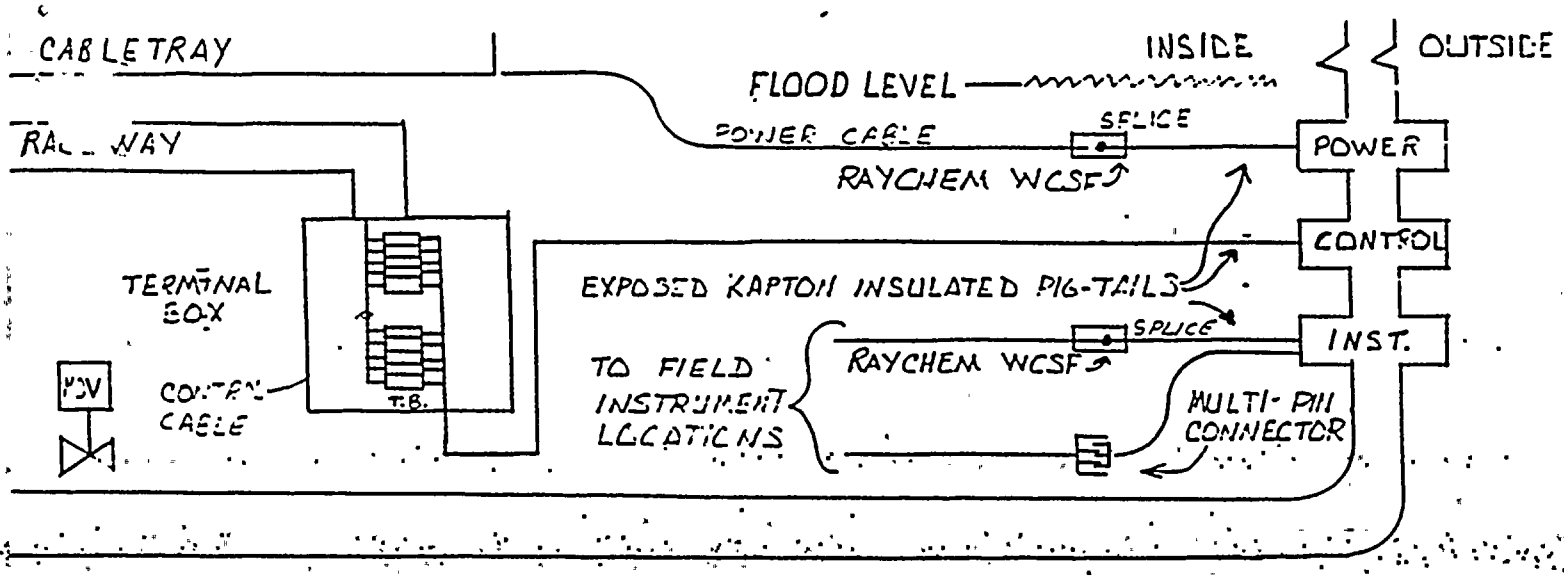

FIG. 1A SHORT TERM

NOTE: Further information regarding this matter is contained in IE Inspection Report 50-316/78-09

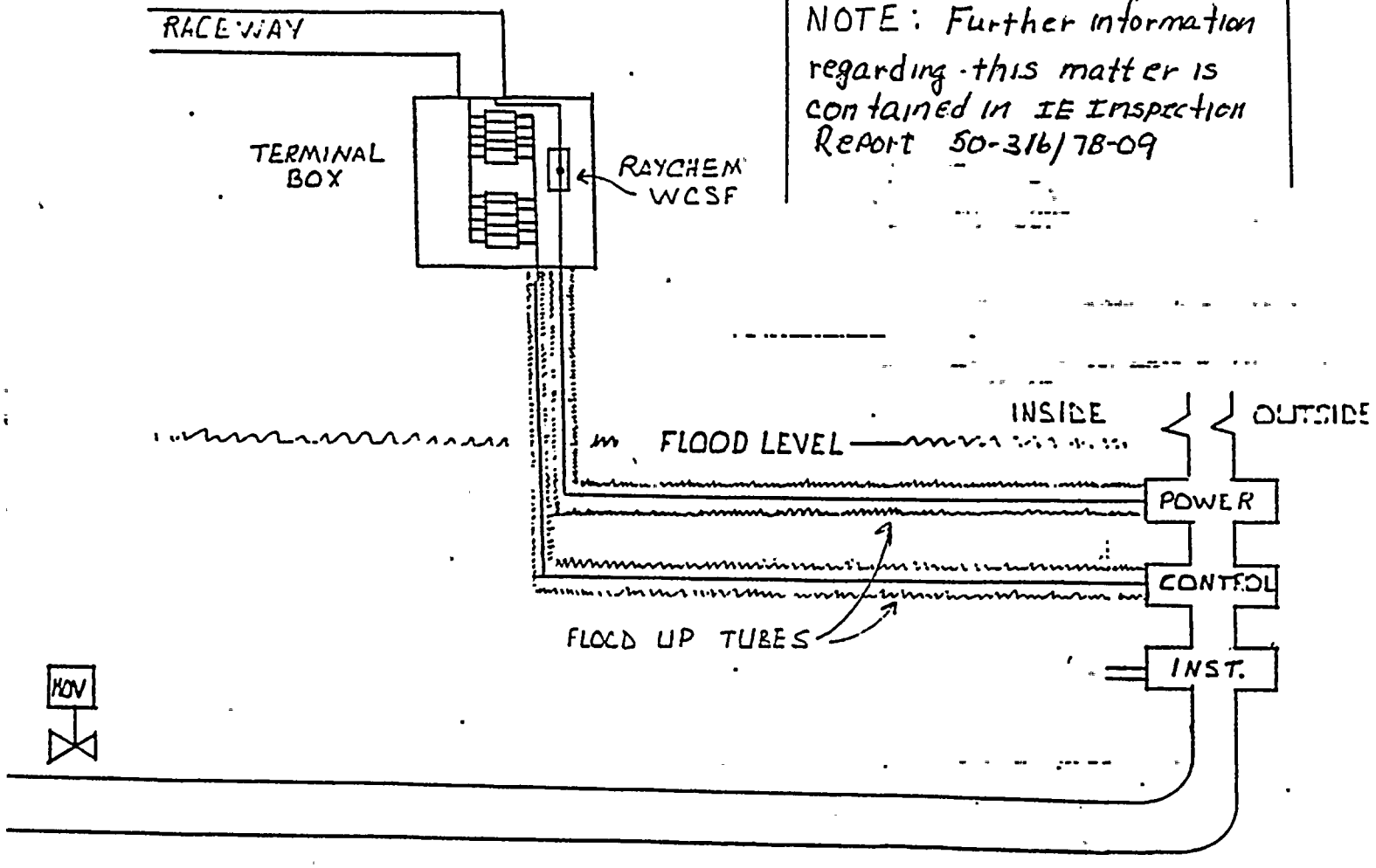

FIG 1B LONG TERM

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

July 23, 1980

MEMORANDUM FOR: E. L. Jordan, Assistant Director, Division of Reactor Operations Inspection, IE:HQ

THRU: G. Fiorelli, Chief, Reactor Construction and Engineering Support Branch

FROM: D. W. Hayes, Chief, Engineering Support Section 2

SUBJECT: IEB 79-01B (A/I F03067180)

Attached is a copy of a memorandum dated July 17, 1980 received from Frank Jablonski relative to IEB 79-01B. It is being forwarded for your information and solicited guidance.

The question of identification of safety related systems and components (paragraph No. 1 of the memo) is an old one. I disagree with Frank in that I feel that this identification is a responsibility of the licensee, not the NRC. He must know his plant. I do agree, however, that more guidance is needed for our inspectors in this area. This is especially important for those inspectors that have not had reactor operating experience.

The significant differences in master lists that Frank discusses in paragraph two does raise questions. We can only compare these lists against the SAR. Review and evaluation beyond this is assumed to be an NRR function.

In regard to Frank's question - should we assume the licensee's response to IEB 79-01B to be complete and correct - I have told him yes. Further, that if he identifies significant incompleteness in the response, or incorrect information during his reviews, to bring these to my attention so appropriate action can be recommended.

Comments and further guidance is requested concerning matters discussed in paragraphs 3 and 4 of Frank's memo.

D. W. Hayes

D. W. Hayes, Chief
Engineering Support Section 2

Generic Issues
ATTACHMENT 3a

Attachment:
F. J. Jablonski Memo to
D.W. Hayes dtd 7/17/80

cc w/attachment:
J. G. Keppler, RIII
V. D. Thomas, IE:HQ
A. Finkel, RI
R. Hardwick, RII
D. McDonald, RIV
J. Elin, RV
R. F. Heishman, RIII
F. J. Jablonski, RIII

ATTACHMENT 3a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

July 17, 1980

MEMORANDUM FOR: D. W. Hayes, Chief, Engineering Support Section 1

FROM: F. J. Jablonski, Reactor Inspector

SUBJECT: FORMULATING TECHNICAL EVALUATION REPORTS (TER) - REVIEW OF IEB 79-01B
RE: MEMO TO YOU DATED JUNE 16, 1980 - SAME SUBJECT

Since the review of IEB 79-01B is continual, new discrepancies continue to show up; discrepancies are not necessarily the licensees'. As you know, there is no specific nuclear power plant design required by NRC. Further, the designation of safety related systems is somewhat arbitrary and inconsistent. In fact, the NRC places responsibility for classifying safety related systems on the licensee.

Action Item No. 1 of 79-01B requested each licensee to provide a "master list" of all ESF systems in their respective plant required to function during a postulated accident. Appendix A to 79-01B lists "typical" equipment/functions needed for mitigation of an accident. A comparison of master lists was made of four licensees with similar Westinghouse PWRs (see Attachment 1). Arbitrary selection and non-standard nomenclature of systems makes evaluation of the master lists extremely difficult. NRC requested each licensee to submit the information under oath. Should the information therefore be assumed complete and correct?

It is extremely frustrating to review responses which vary so much in attention to detail, depth of review, etc. As stated previously in the draft TER for D.C. Cook, because I as a principal reviewer lack detailed systems/operations experience, further guidance is requested.

Another TER related matter is motorized valves equipped with Limitorque operators (see Attachment 2). As can be seen, each test report is for a specific unit type including motor type and insulation class. Almost all licensees refer to the various test reports as qualification documentation for all series of operator types; never is name plate data provided. For example, test report No. 600456 (SMB-0-40, Reliance Motor with Class RH insulation) may be listed for all operators from series SMB-000 to SMB-5; motor name plate data not provided. Without the name plate data and the basis for extrapolation, a meaningful evaluation cannot be made.

ATTACHMENT 3a

It is requested that this memorandum be forwarded to IE:HQS as an addition to A/I F03067180 with the same copy distribution.

F. J. Jablonski

F. J. Jablonski
Reactor Inspector

Attachments:
1. Comparison of Master Lists
2. Motor Operated Valve Tests

cc:
J. G. Keppler
G. Fiorelli

ATTACHMENT 3a

ATTACHMENT 1

SYSTEMS	P.I.	COOK	KEW.	PT. BCH.
Aux. F.W.	X	X		X
Chem. & Vol. Cont.	X	2	X	X
Cntmt. Air Hndlg.	X	X		X
Cntmt. H_2 Cont.	X	X		
Cntmt. Sp.	X	X		1
Main Stm.	X	X		X
Aux. Stm.	X			
Stm. Dump	X			
Rx Clnt.	X	X	X	X
Res. Ht. Rm. [1]	X	2	X	3
Saf. Inj. [1]	X	2	X	X
Clg. Water	X			
Esnt'l. Serv. Wat.		X		
Comp. Clg. Wat.		X		3
Emerg. Core Clg. [2]	1	X	1	
Aux. Clnt. [3]				X
Cntmt. Purge	X			
Rx. Bldg. Vent			X	
Inst. & Prot.	X			
Rx. Trip. Act.		X		
Rx. Cont. & Prot.				X
Rad. Monit.	X			
Rx. Hot Samp.	X			
Stn. & Inst. Air	X			
Stm. Gen.BD	X			
Post Acc. Monit.		X		
Rem. Sht. dn. Monit.		X		
Cntmt. Isol.		X		X
Mn. Stm. Isol.		X		
Mn. FW Isol.		X		

ATTACHMENT 2

MOTOR OPERATED VALVES

MOV's

1. There are basically two type series of Limitorque operators: SMB and SB. The operators are sized from 000 (smallest) to 5 (largest) as follows:

SMB-000
SMB-00
SMB/SB-0[1]
SMB/SB-1
SMB/SB-2
SMB/SB-3
SMB/SB-4
SMB-5

This series may also include SB

This series may also include WB

This series may be suffixed "T"

2. Test Reports include:

	Report No.	Date	Unit Type	Environment	Motor Type	Insulation
a.	600198	1-2-69	SMB-0-15*	PWR No Radiation	Reliance	Special Hi Temp
b.	600426 (B-0009)	4-30-76	SMB-0-25*	BWR 1×10^7 R 340°	Peerless DC	H
c.	600376A FIRL F-C 3441	5-15-76	SMB-0-25*	BWR 2×10^8	Reliance	RH
d.	600456	12-9-75	SMB-0-40*	PWR 2×10^8	Reliance	RH
e.	600461	6-7-76	SMB-0-25*	Outside Cntmt, 2×10^7	Reliance	B
f.	WCAP7410L 7744	12-70 8-71	SMB-00			B

*denotes foot pounds of torque

[1] only SMB-0 has been tested seismically Re: a, b, c

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SSINS #6820

JUL 3 1980

MEMORANDUM FOR: Z. R. Rosztoczy, Branch Chief, Equipment Qualification Branch, Division of Engineering, NRR

THRU: Haw for E. L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operations Inspection, IE

FROM: V. D. Thomas, Task Manager, Review Group, IEB 79-01B, Division of Reactor Operations Inspection, IE

SUBJECT: REQUEST FOR NRC POSITIONS ON REVIEW QUESTIONS OF IEB-79-01B LICENSEE RESPONSES

In accordance to our verbal agreement, we would be happy if you would provide positions on the questions noted in the enclosed memoranda.

Since it is essential to establish a uniform approach to the review effort to obviate the questions being generated in the on-going review of licensee responses, we will be happy to meet with your staff to discuss these concerns to expedite resolution of the issues.

Vincent D Thomas

Vincent D. Thomas, Task Manager
Review Group, IEB 79-01B

Enclosures:
1. Memo D. W. Hayes to G. Fiorelli, RIII dated June 20, 1980.
2. Memo F. Jablonski to D. Hayes, RIII dated Jun 16, 1980.
3. Memo F. Jablonski to D. Hayes, RIII DATED June 10, 1980.

cc: w/enclosures
E. L. Jordan, IE
V. S. Noonan, NRR
G. Fiorelli, RIII
D. W. Hayes, RIII
A. Finkel, RI
R. Hardwick, RII
F. Jablonski, RIII
D. McDonald, RIV
J. Elin, RV

JUL 7 1980

ATTACHMENT 3a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

June 20, 1980

MEMORANDUM FOR: E. L. Jordan, Assistant Director, Division of Reactor Operations Inspection, IE:HQ

THRU: G. Fiorelli, Chief, Reactor Construction and Engineering Support Branch

FROM: D. W. Hayes, Chief, Engineering Support Section 1

SUBJECT: IEB 79-01B (A/I F03067180)

Attached are two memorandums from one of my inspectors, Frank Jablonski. The first is dated June 10, 1980 and the second June 16, 1980. Both memos raise basic questions for which we require guidance to complete our review of responses to IEB 79-01B.

By this memo I also would like to confirm our understanding that NRR (Environmental Qualification Branch) will review for acceptability all test reports and other documentation which licensees reference as establishing environmental qualification of instrument/electrical equipment. In connection with this, we are sending under separate cover test reports, etc. in our possession to be forwarded to the Environmental Qualification Branch. (We further understand that the IEB 79-01B task group, on a volunteer basis, may agree to review some of these documents).

The status or schedule for site inspections and review/evaluation of the final reports is also attached. Please note that every licensee has asked for some sort of time extension to submit their first report. We understand that the other regions have had similar reporting problems. Assuming that all our licensees meet their extended submittal dates, we should complete our site inspections, reviews, and technical evaluation

ATTACHMENT 3a

reports by the end of December 1980. Further delays in the submittals or any unforeseen events will hamper our ability to meet the new February 1, 1981 deadline.

D. W. Hayes, Chief
Engineering Support Section 1

Attachments:
1. Memo F. Jablonski to D. Hayes 6/10/80
2. Memo F. Jablonski to D. Hayes 6/16/80
3. Inspection Status/Schedule
4. "Separate Cover" List (Test Reports Sent to IE:HQ)

- Separate Cover: See Attachment 4

cc w/attachments 1, 3, & 4 only:
J. G. Keppler
G. Fiorelli
V. D. Thomas, IE:HQ
A. Finkel, RI
R. Hardwick, RII
D. McDonald, RIV
J. Elin, RV
R. F. Heishman

ATTACHMENT 3a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

June 10, 1980

MEMORANDUM FOR: D. W. Hayes, Chief, Engineering Support Section 1

FROM: F. J. Jablonski, Reactor Inspector

SUBJECT: EFFECT OF PREVIOUS NRR REVIEW ON MATTERS RELATING TO IEB 79-01B

In almost every licensee response to IEB 79-01B there is a subtle or direct reference to matters apparently reviewed by NRR. Because of the referenced dates it is assumed by me that NRR has given either tacit or direct approval to the references; examples follow:

1. All licensees refer to their FSARs for establishing the list of engineered safety feature systems and environmental data such as temperature, pressure, radiation, etc.

2. One licensee, Wisconsin Public Service Corporation, states that "The AEC, in their "Safety Evaluation of the Kewaunee Plant", Section 7.5, issued July 24, 1972, concluded that our criteria and testing program for environmental qualification were adequate". It is further stated that "Our FSAR, which was approved by the AEC, discusses at length the post accident conditions and required qualifications for applicable equipment. (See Section 7.5 of the Kewaunee FSAR.)"

3. Two licensees, American Electric Power and Wisconsin Public Service Corporation, have discussed the effect of components below flood level simply by referencing letters previously submitted to the NRC, or FSAR questions/answers as follows:

   * AEP - Letter dated 9-29-75 from Tillinghast (AEP) to Kniel (NRC); FSAR question 40.10 Appendix Q.

   * WPSC - Letter dated 2-2-76 from James (WPSC) to Purple (NRC).

ATTACHMENT 3a

My specific concerns are:

Is it to be assumed that the referenced FSAR parameters, No. 1 above, are correct, i.e. reviewed by NRR?

If the answer is yes, then should it also be assumed that No. 2 above is likewise adequate? (If the answer is no, then none of the licensee responses which reference the FSAR can be assumed to be correct.)

Reference No. 3, even though a component may not be required to operate subsequent to flooding, what effect will short circuits have on containment electrical penetrations? Was this considered by NRR?

I am requesting that these questions/concerns be forwarded to the Assistant Director, Division of Reactor Operations Inspection for resolution.

F. J. Jablonski

F. J. Jablonski
Reactor Inspector

cc:
J. G. Keppler
G. Fiorelli

ATTACHMENT 3a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

June 16, 1980

MEMORANDUM FOR: D. W. Hayes, Chief, Engineering Support Section 1

FROM: F. J. Jablonski, Reactor Inspector

SUBJECT: FORMULATING TECHNICAL EVALUATION REPORTS (TER) - REVIEW OF IEB 79-01B

In accordance with IEB 79-01B, an overall conclusion relative to the qualification of instrument electrical equipment is to be made for each operating plant based on a screening review of all plant systems, and by a detailed review and observation of specific system components. Unresolved concerns previously identified by RIII inspectors during reviews of IEC 78-08 and IEB 79-01 along with subsequently identified concerns make it difficult for us to formulate meaningful TERs for certain plants. The previous unresolved concerns are documented in the memorandums listed below (1,2,3) and are reiterated in Attachment A to this memo. Subsequently identified concerns are listed in Attachments B, C, and D.

To assure uniform evaluation, guidance is needed for these items. Please forward these concerns to IE:HQ.

1. TI 2515/13 - Qualification of Safety Related Electrical Equipment Fiorelli to Sniezek, 10/13/78

2. Same title as 1., Fiorelli to Klinger, 12/78

3. Review Status of Responses to IEB 79-01, Hayes to Jordan, 9/5/79

F. J. Jablonski

F. J. Jablonski
Reactor Inspector

Enclosures: As Stated

cc:
J. G. Keppler
G. Fiorelli
V. D. Thomas, IE:HQ
A. Finkel, RI
R. Hardwick, RII
D. McDonald, RIV
J. Elin, RV

ATTACHMENT 3a

ATTACHMENT A

1. Foxboro Models EIIGM and 611/613 transmitters with MCA modification are believed by RIII to be under a generic review by NRR. It is RIII's further belief that the "MCA" modification does not make the transmitters suitable for use in a radiation environment. Is Region III's understanding correct?

2. Several licensees have declined replacement of limit switches which provide position indication of valves used for primary containment isolation. Are these switches required to be qualified?

3. GE cable type SI-57275 is used on penetrations manufactured by GE. Penetrations with this type cable are installed at Monticello, Dresden 1 and 2, Quad Cities 1 and 2, and Duane Arnold. The cables withstood LOCA tests performed by Wyle Laboratories, Report No. 44114-2; however, the cable did not pass the IPCEA S-19-81 vertical flame test. Further, in the same Wyle test, GE cable SI-58136 failed at radiation levels in excess of 5×10^6 rads. We recognize that in regard to GE cable type SI-57275 flame tests are not part of the environmental qualifications addressed in IEB 79-01B, but it makes no sense to find these penetrations acceptable per IEB 79-01B knowing that they may not meet other requirements.

   Concerning GE type SI-58136 cable, this item should be evaluated on a generic basis since many of the early GE plants use this cable.

4. One licensee, American Electric Power, lists a letter No. NS-TMA-1950, W to NRR, as technical reference for qualification of ITT Barton differential pressure transmitters. Please supply us with the disposition and status of the letter.

ATTACHMENT B

The following questions are based on our review of some licensee submittals to IEB 79-01B:

1. Licensees maintain that aging is not a required consideration for components that are included in a routine periodic inspection and calibration program. Is this acceptable?

2. Licensees maintain that aging is a generic industry issue whose resolution is not clear; therefore, evaluation has not been made or will continue to be made as relevant information is made available.

3. Licensees are referencing manufacturers' letters as establishing the qualification of ancillary parts such as lubricants, tapes, etc. Is this acceptable or are manufacturers' test reports required?

4. Limit switches used for valve position indication only have been deleted from the submittal. Licensees maintain that a valve outside containment in series with one inside can have its position verified visually following an accident. Is this acceptable?

5. The licensees maintain that neither valve position limit switches, solenoid valves, nor control cables for air-operated containment isolation valves need be replaced or protected from the adverse environment, including flood, because all postulated failures will result in the isolation valve assuming its fail-safe position. Is this acceptable.

6. Some fan cooler motors do not meet FSAR requirement of 1.5×10^8 rads. Qualification test was to 1.4×10^8 rads. Licensee states radiation

level is "close enough" to expected accident radiation level to be acceptable. Is this acceptable to the NRC?

7. Attachment D is a summary of problems incurred during a one year operation test of a containment fan cooler unit. Would you consider the test to be a success?

ATTACHMENT C

1. In lieu of a test report, what constitutes an acceptable Certificate of Compliance?

2. What if the test specimen and installed component differ, e.g. model, type, etc?

3. What, as a minimum, must be included for an analysis to be acceptable?

4. The guidance provided in Enclosure 4 of 79-01B allows analysis (evaluation) for service conditions such as radiation and chemical sprays. Is analysis (evaluation) and "engineering judgment" the same thing?

5. Since effects of radiation and chemical spray are "allowed" to be analyzed (evaluated) for important components such as containment electrical penetrations, is it prudent to require a licensee to prepare a full blown analysis to qualify a 7/C 12 AWG cable when a similar 5/C 14 AWG cable was actually tested and shown to be qualified?

6. Provide us with ± limits for evaluation of test data such as pressure, temperature, radiation, duration, chemical spray, and aging.

7. Most tests include only single components and the reports do not include any acceptance criteria. Test conclusions are that usually, no matter what happens during the test, the component is accepted. This is commonly referred to as a "dead bug" test. Provide us with minimum acceptance criteria requirements for a test and its report to be acceptable.

ATTACHMENT D

[illegible] CO.
[illegible], CALIFORNIA

PAGE 1 OF _____
REPORT NO X-F-1
PREPARED BY [illegible]
CHECKED BY J.T. [illegible]
DATE April [illegible], 1977

The following is a brief summary of the problems incurred during the one year extended operation phase of the qualification test.

DATE	TIME ON [illegible]	LENGTH OF [illegible] TIME	REASON
12-13-75	75.3	1-3/4 Hours	Maintenance problems caused loss of power.
12-17-75	100.2	2-1/4 hours	Transformer coil burned out.
12-25-75	373.8	2-1/4 hours	Loss of plant power.
1-13-76	827.5	15 min.	Electrical storm caused loss of plant power.
2-17-76	1648	9 hours	Spray rings plugged. Rigged bypass for cooling water. Shutdown required to drill holes in test chamber.
2-21-76	1741.5	15 min.	Plant maintenance required cut-off of power.
3-3-76	2211.5	40 min.	Electrical storm caused two short shutdowns because of temporary power interruption.
3-18-76	2352.7	4 hours	Tripped on overload. Faulty solenoid caused condensate to back up in chamber.
3-19-76	2384	2 hours	Unknown.
4-12-76	2047.5	10 days	Bearing problem; see Appendix C.
4-22-76	2384	5-1/2 hours	Unknown.
4-29-76	3007.9	9 hours	Unknown - Installed recording equipment amps and voltage to detect nuisance trips.
4-30-76	3022.6	2-1/2 days	Nuisance trip due to overload heater failure; length of shutdown because failure occurred late Friday and parts not available until Monday.

ATTACHMENT 3a

DATE	TIME ON HOUR METER	LENGTH OF DOWN TIME	REASON
5-9-76	5140	No Trip	Problem with dump valves but corrected without shutdown.
7-15-76	4759.5	1 hour.	Power failure due to electrical storm.
7-30-76	5109.8	2-1/2 days.	Terminal board ruptured causing loss of pressure in chamber. This board was a seal required for testing and was not a part of equipment being qualified.
8-14-76	5369.2	6 hours	Power failure.
8-19-76	5487.3	No Trip	Slight problem in the controls causing a slight cycling of temperature. Problem corrected without shutdown.
8-19-76	5502.6	6-1/2 hours	Faulty solenoid resulting in condensate backing up in chamber and causing motor to trip on overload.
8-20-76	5601.4	2-1/2 hours	Solenoid did not function properly and override circuit did not operate. Motor tripped on overload.
8-30-76	5725	No Trip	Temperature was down slightly. Solenoid had failed to operate but override circuit was venting chamber. Problem was corrected without shutdown.
8-31-76	5752	2 hours	Motor tripped on overload. Float valve stuck causing condensate to build up in chamber.

ATTACHMENT 3a

JOY MANUFACTURING CO.
NEW PHILADELPHIA, OHIO

REPORT NO [illegible]
PREPARED BY T.A. [illegible]
CHECKED BY J.T. [illegible]
DATE April [illegible], 1977

DATE	TIME ON HOUR METER	LENGTH OF DOWN TIME	REASON
9-8-76	5954.7	1-1/2 days	Breakdown in electrical tape and other insulation applied by JOY at the ends of the lead caused a short which burned off one lead resulting in a single phased condition. Lead end was repaired.
11-9-76	7384.3	2 days	Lead separation at terminal board caused by breakdown of insulation, lead embrittlement and vibration. Lead was repaired.
11-30-76	7840.5	20 hours	Insulation failure similar to that of 9-8-76 caused unit to trip.
12-24-76	8390.9	1 hour	Improperly assembled solenoids caused condensate to back up in chamber resulting in an overload trip.
12-30-76	8530	No Trip	Lost Phase #1; continued to operate in a single phase condition.
1-11-77	8814.2	27 hours	Power surge caused unit to trip. Because of single phase condition unit could not be restarted. Repaired the lead at the terminal board and resumed testing. This break was similar to that of 11-9-76.
3-9-77	10145.9	--	Apparent short in motor caused termination of test.

ATTACHMENT 3a

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

July 10, 1980

MEMORANDUM FOR: E. L. Jordan, Assistant Director, Division of Reactor Operations Inspection, IE:HQ

THRU: G. Fiorelli, Chief, Reactor Construction and Engineering Support Branch

FROM: D. W. Hayes, Chief, Engineering Support Section 1

SUBJECT: FINAL REVIEW OF LICENSEE RESPONSE TO IE 79-01B - DRAFT TECHNICAL EVALUATION REPORT FOR D.C. COOK UNITS 1 AND 2 - DOCKET NOS. 50-315; 50-316

Attached to this memorandum is the draft technical evaluation report (TER) for the D.C. Cook plant. The format is submitted for comment by the task group; the content is submitted for review and resolution by NRR.

D. W. Hayes, Chief
Engineering Support Section 1

Attachment: As Stated

cc: w/attachment
J. G. Keppler
G. Fiorelli
R. Heishman
Resident Insp.
V. Thomas, IE:HQ
A. Finkel, RI
R. Hardwick, RII
D. McDonald, RIV
J. Elin, RV

Specific Issues
ATTACHMENT 3b

DRAFT TECHNICAL EVALUATION REPORT OF THE ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT INSTALLED AT THE D.C. COOK NUCLEAR POWER PLANT - RE: IEB 79-01B

INTRODUCTION

This Technical Evaluation Report (TER) is the result of reviews and inspection necessitated by IEB 79-01B; however, it is also the culmination of IE inspection activities which started in 1975. Attachment 1 includes a synopsis of specific IE matters relating to the development of this TER.

In addition to IE activities relative to environmental qualification, NRR has also accumulated several years of past experience, expended resources, performed reviews, held meetings and various telephone conversations, issued and made amendments to licenses, etc. It is for these reasons that NRR performed the detailed review of licensee submitted test/qualification documentation.

This TER does not include information about seismic or fire withstand capability. It should therefore not be inferred that equipment which is "environmentally" qualified meets all necessary qualification requirements. It should be noted that environmental data may be extrapolated directly to similar materials; however, because of significant differences in physical size, direct correlation may not be justified for seismic considerations.

Purpose

The purpose of this TER is to document IE's role and effort in the matter of IEB 79-01B, and forward the results to NRR's Equipment Qualification Branch for their use in developing the Safety Evaluation Report (SER) for the D.C. Cook plant.

Method

On June 5, 1980 the licensee submitted the final response to IEB 79-01B describing in detail their review of equipment used to mitigate the consequence of an accident. Equipment installed in potentially harsh environments was of primary interest.

On June 16-17, 1980, IE-RIII performed a site inspection of specific system components in order to verify the accuracy of the licensee's response. (Refer to Attachment 1, item B5). Further, a complete review of the submittal was made to identify discrepancies and determine if the submittal references actually supported equipment qualifications. (Refer to IE:RIII Review Results below).

IE:HQ and NRR were to review and evaluate IE-RIII inspection results and other potential problems, provide guidance for uniform evaluation of licensee submittals, perform detailed reviews of all licensee referenced test/qualification documentation submitted by IE:RIII. (Refer to Attachment 1, Items B1-4). NOTE: No response has been received by IE:RIII

to the above referenced inquiries.

Licensee Results

In the response letter, the licensee stated the list of equipment was complete and proof of qualification to withstand adverse environment existed in all cases except where noted in the letter, attachment 4, 5, and 6 to the response, and in various worksheets. A synopsis of those exceptions follows:

*Aging is a generic industry issue whose resolution is not clear at this time. As relevant information becomes available, the licensee intends to continue its evaluation of the sensitivity to aging of materials and components.

*Limit switches used only for indication of valve position will not be replaced.

*Components for the pressurizer relief valves, including solenoid and motorized valves, limit switches, and terminations are being considered for upgrade as a result of TMI lessons learned.

*Instrument cable is located below flood level for several instruments.

*Control cable is located below flood level for several containment

isolation valves.

*Auxiliary feedwater flow transmitters located outside containment will be replaced.

*Emergency service water pressure swiches located outside containment will be replaced.

*Motorized valves and terminations in the RHR System may not be qualified for exposure to radiation. (Outstanding item).

IE:RIII Review Results

In addition to the discrepancies noted during the site inspection, the following potential discrepancies were noted during review of the entire submittal. Due to lack of guidance from NRR's Environmental Qualification Branch and lack of detailed systems/operations experience of the principal reviewer, resolution was not possible.

*Continental instrument cable failed a LOCA test as documented in Conax test report IPS-348. Continental control cable, made of what is believed to be the same insulating and jacketing material, "passed" the same test. Because only one sample of each specimen was tested and no real acceptance criteria except "survival" was used, it is the principal reviewer's opinion that the Continental control cable should not have been deemed acceptable. Continental control cable is installed inside containment

and below flood level.

*No correlation can be made between the test reports and instrument terminations on submittal page TI-1-1, TI-2-1, and TI-4-1. None of the tests included submergence, yet the worksheets indicate terminations are qualified for submergence. The terminations are below flood level.

*Worksheets TI-3-1 and TP-1-1 infer all penetration terminations are made in flood up tubes; not true.

*Penetration terminations for motorized valves, hydrogen re-combiner and fan motors are located below flood level. (Worksheet TP-3-1).

*Core cooling and containment isolation valves are located below flood level. (Worksheet VI-1).

*Long term monitoring steam generator instruments are located below flood level. (Worksheet I).

*Long term monitoring boron injection tank discharge flow transmitters are located below flood level; referenced by emergency procedures. (worksheets I-7-8).

*None of the referenced lubricants/greases had been "qualified" for LOCA by test. Letters from suppliers and/or manufacturer's specifications were

referenced. (Worksheets G1-G11).

*In many cases, the component worksheet specified operating time is one year; in only limited cases has that time been qualified.

Summary

Licensee submitted response to IEB 79-01B. IE:RIII performed review of submittal, made site inspection and forwarded to NRR via IE:HQS inspection results and requests for specific guidance.

Conclusion

It cannot be concluded, at this time, that there is reasonable assurance all components installed at the D.C. Cook nuclear power plant are environmentally qualified, nor can it be concluded that installation methods of environmentally qualified components would not contribute to the failure of such components during a potential accident. A positive conclusion cannot be made until all matters referred to IE:HQS/NRR have been satisfied, and NRR has reported the results of their review of all test reports.

Principal Reviewer

ATTACHMENT I

MATTERS RELATING TO DEVELOPMENT OF THE TER FOR D.C. COOK

A. IE INSPECTION REPORTS

1. 50-316/75-04
2. 50-316/75-07
3. 50-316/76-01
4. 50-316/76-04
5. 50-316/77-09
6. 50-316/77-38
7. 50-315/78-08; 50-316/78-03
8. 50-316/78-09
9. 50-315/78-24; 50-316/78-22
10. 50-315/78-31; 50-316/78-29
11. 50-315/79-16; 50-316/79-09

B. MEMORANDA

1. 12/28/77 Heishman to Reinmuth
2. 1/12/78 Heishman to Sniezek
3. 1/16/78 Sniezek to Kniel
4. 6/20/80 Hayes to Jordan
5. 7/1/80 Hayes to Jordan

C. AMENDMENT 4 TO LICENSE DPR-74

D. DRAFT TECHNICAL EVALUATION REPORT OF THE ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT INSTALLED AT THE D.C. COOK NUCLEAR POWER PLANT.

1. Auxiliary Feed Water *
2. Chemical and Volume Control **
3. Containment Air Handling
4. Containment H_2 Control
5. Containment Spray
6. Main Steam
7. Reactor Coolant #
8. Residual Heat Removal *
9. Safety Injection *
10. Essential Service Water
11. Component Cooling Water
12. Emergency Core Cooling
13. Reactor Trip Actuation
14. Post Accident Monitoring
15. Remote Shut Down Monitoring
16. Containment Isolation
17. Main Steam Isolation
18. Main Feedwater Isolation
19. Engineered Safeguards Actuation
20. Main Feedwater Control

*Emergency Core Cooling
#Pressurizer and Relief Valves
**Containment Radiation Monitor

Licensee's System List
ATTACHMENT 4

SYSTEMS LIST

(W) PWR

1. Reactor Protection
2. Safeguards Actuation
3. Containment Isolation
4. Main and Auxiliary Steam Isolation
5. Main and Auxiliary Feedwater Isolation
6. Feedwater Control
7. Containment Spray
8. Containment Air Purification/Cleanup
9. Containment Ventilation/Cooling
10. Containment Combustible Gas Control
11. High Pressure Injection
12. Low Pressure Injection
13. Accumulators
14. Residual Heat Removal
15. Chemical and Volume Control
16. Pressurizer Spray
17. Power Operated Relief Valves
18. Steam Dump
19. Containment Radiation/Monitor
20. Containment Radiation/Sample
21. Component Cooling Water
22. Service Water
23. Emergency Power
24. Containment Sump
25. Control Room Habitability
26. Safety Equipment Area Ventilation
27. Auxiliary Feedwater
28. Reactor Coolant

CATEGORY

I. Equipment is Qualified for Plant Life

a. Equipment meets all applicable requirements of DOR Guidelines or NUREG-0588.

b. Qualification by judgement may be acceptable with sufficient basis.

II. Equipment is Qualified with Restrictions

Equipment meets all applicable requirements of DOR Guidelines or NUREG-0588 with the following limitations:

a. Equipment Qualification for service life less than the plant life.

b. Equipment requires modification to meet qualification requirements, such as relocation or shielding.

III. Equipment is Exempted from Qualification

Equipment where safety related function can be accomplished by redundant fully qualified equipment which meets single failure criteria.

IV. Qualification of Equipment Unresolved

a. Qualification resting scheduled, but not complete.

b. Qualification Records search still in progress.

V. Equipment Not Qualified

LER

LER's

None

LER'S
ATTACHMENT 7

UNRESOLVED GENERIC - SPECIFIC ISSUES

1. No answer was ever recieved to the Generic Issues, Attachment 3a, discussed in Attachment 2 of memorandum Hayes to Jordan dated June 20, 1980.

2. No answer was ever received to the Specific Issues, Attachment 3b, discussed in the attachment of memorandum Hayes to Jordan dated July 10, 1980.

Unresolved Generic - Specific Issues
ATTACHMENT 8

A. Pending review and approval by NRR/EQB of test reports or other documentation.

B. Reviewer not qualified.

C. Test specimen different from installed equipment. ***

D. To be replaced; when not specified.

E. Fails Safe.

F. Operates before flood. Effect of short circuits on containment electrical penetrations not considered.*

G. Analysis - Engineering review.

H. Limit switches used for valve position only.

I. Long term testing not performed. ** 1, 2

J. Ancilliary components being considered for upgrade. (NUREG-0578)

K. Calculated containment temperature 2.78 hours (10,000 seconds) after a LOCA is 185°F and decreasing. The electrical cable is rated for continuous operation at 194°F (90°C). Therefore, the containment environmental temperature after 2.78 hours does not represent a challenge to the mechanical or electrical quality of the cable. (Margin of 9°F; IEEE 323 is 15°F)

L. 230°F for ten seconds and 11.5 psig for 0.1 seconds will not challenge the mechanical or electrical quality of the termination.

M. No details on aging.

N. Aging: 250°F/7 days; 300°F/54 days; 390°F/20 days; 302°F/7 days; 360°F/4 days; 165°F/8 days; 200°F/8 days. Which one is acceptable?

O. Referenced by emergency procedures.

P. Materials listed in Table C-1 of IEB 79-01B indicate a potential for significant thermal aging when exposed to normal operating conditions. (XLPE/Hypalon 40 years; EPR/Neoprene 10 years)

Q. Not included in routine replacement program.

R. Radiation specification determined by AEPSC NS&L calculation DC-N-6420-2. (Not reviewed.)

S. Shielded; not in direct line of stated accident.

*Letter Tillinghaust to Kniel dated September 29, 1975; FSAR Appendix Q questions 40.6, 40.10, and 40.12. (Attached)

Concurrence Code
ATTACHMENT 9

T. No credit for assumed operation.

U. Limited test length.

V. Main steam line break plus failure of another steam line to isolate is acceptable. Two steam generator stop valves can be affected by any one break.

W. Unable to evaluate; nine test reports listed.

X. Even though beyond the scope of IEB 79-01B, it is unclear whether in Unit 1 justification has been made for use of cables which did not pass the IPCEA-19-81 vertical flame resistance test. Reference: November 25, 1977 NRR letter to licensee, subject: "Summary of Meetings Held on November 3 and 4, 1977, Reference - Fire Protection". (Question 49).

Y. Below flood level.*

Z. Generic Issue

* The issue of possible short circuit interactions with safety related equipment including potential degradation is beyond the 79-01B review scope. However, this issue was raised as part of the Unit 2 license review under Reg. Guide 1.63 conformance (Q40.6) (a copy of Q40.6 and its response are also included for your information and use), and AEP responded to this issue by installing redundant circuit breakers for all equipment inside containment (both safety and non-safety-related) to alleviate this sort of problem. Installation of this system was a Unit 2 license condition and is 100% installed in that Unit. We are also installing the same protection in Unit 1 even though no committment was made to the NRC to do so. The work in Unit 1 is 95% complete.

**1 The penetrations which are specified for long term operation are qualified for more than one year's expected radiation dose. In addition the penetrations have been tested for submergence (test ran as long as 6 days) and performed properly after the testing. Documentation of this can be found on pages 8 and A5 of IPS-326 and pages 4 and A4 of IPS-327.

**2 The subject WCAP actually contains two tests. A "short term" test was run in which the fan motor was operated without its associated heat exchanger which provides component cooling. The second "long term" test involved operation of the motor with the heat exchanger operable as it would be during the long term post-accident configuration. We Westinghouse has determined that the long term post-accident operation of the motor is qualified for indefinite long term operation. In our upcoming submittal, the qualification chart for this fan motor (F1) will be formally updated to reflect this position.

ATTACHMENT 9

*** The fan motor in question was carefully identified by AEP. As per our request, the supplier (Westinghouse) has explicitly stated that this fan motor is qualified by the subject report. (A copy of the letter documenting this qualification (AEP-79-583, Noon to Caso) is also attached for your information).

List A

- Engineered Safeguards Actuation
  Containment Phase A Isolation Actuation
  Main Steam Isolation Actuation
  Reactor Trip Actuation
  Main Steam Normal System Monitoring

List B

- Engineered Safeguards Actuation
  Containment Phase A Isolation Actuation
  Post Accident Monitoring
  Reactor Trip Actuation
  Main Steam Normal System Monitoring
  Remote Shut-down Monitoring

List C

BLANK

List D

- Engineered Safeguards Actuation
  Containment Phase A Isolation Actuation
  Main Steam Isolation Actuation
  Main Feedwater Isolation Actuation
  Reactor Trip Actuation
  Reactor Coolant Normal System Monitoring

List E

- Engineered Safeguards Actuation
  Containment Phase A Isolation Actuation
  Remote Shut-down Monitoring
  Reactor Coolant Normal System Monitoring

ATTACHMENT 9

List F

Engineered Safeguards Actuation
Containment Phase A Isolation Actuation
Post Accident Monitoring
Containment Phase B Isolation Actuation
Containment Spray Actuation
Main Steam Isolation Actuation

List G

Post Accident Monitoring
Containment Phase B Isolation Actuation
Containment Spray Actuation
Main Steam Isolation Actuation

List H

Post Accident Monitoring
Main Feedwater Isolation Actuation
Reactor Trip Actuation
Main Feedwater Normal System Monitoring

List J

Reactor Trip Actuation
Post Accident Monitoring
Remote Shut-dwon Monitoring
Reactor Coolant Normal System Monitoring

ATTACHMENT 9

Westinghouse Electric Corporation
Water Reactor Divisions
Nuclear Service Division
Box 2728
Pittsburgh Pennsylvania 15230

AEP-79-583

July 3, 1979

Mr. L. F. Caso
Electrical Generation Section
American Electric Power Service Corporation
2 Broadway
New York, New York 10004

Dear Mr. Caso:

Subject: Containment Air Recirculation Fan Motor Environmental Qualification

Reference: AEP-5479 dated March 30, 1979 and AEP-5501 dated June 18, 1979

This letter is to advise American Electric Power that the containment air recirculation fan motors installed in D. C. Cook Units 1 and 2 have been environmentally qualified by Westinghouse.

The environmental qualification of these motors is documented in the following Westinghouse WCAPs, topical reports, and drawing:

1) WCAP-7829 "Fan Cooler Motor Unit Test"
2) WCAP-7343-L "Irradiation Testing of Reactor Containment Fan Cooler Motor Insulation"
3) WCAP-7396-L "Safety Related Research and Development for Westinghouse Pressurized Water Reactors, A Program Outline Fall 1969"
4) WCAP-7498-L "Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries Spring 1970"
5) WCAP-7722 "Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries Spring 1971"
6) WCAP-9003 "Fan Cooler Motor Unit Test"
7) Westinghouse Drawing 206C391 "Safeguards Containment Motor Cable Connector Insulation"
8) Containment Air Recirculation Fan Motor Seismic Analysis

WCAP-7829 was transmitted to Ms. Joan Etzweiler on April 9, 1979 by my letter AEP-79-539. WCAPs-7343-L, 7396-L, 7498-L, 7722, 9003, and Westinghouse drawing 206C391 were transmitted to Ms. Joan Etzweiler on May 4, 1979 by my letter AEP-79-551. The fan motor seismic analysis was transmitted to you on May 8, 1979 by my letter AEP-79-555.

Please be aware that Westinghouse drawing 206C391 provides details on the method of connection of power to the motor qualified by Westinghouse and does not necessarily apply to the connections at D. C. Cook. It is AEP's responsibility to ensure that the method of connection and the insulation materials utilized at the D. C. Cook units is equal to or better than that shown on drawing 206C391.

The information contained in the subject WCAPs, topical reports, and drawing will provide documentation of Westinghouse qualification testing for seismic, SLB and LOCA. Also included are the lubricant used and its condition after the test; the Westinghouse method of connecting the motor leads to the power cables, the insulating materials used by Westinghouse on the motor pigtails with details of the motor qualified in a simulated accident test.

I trust that this information will allow you to respond to I&E Bulletin 79-01.

Sincerely,

F. Noon, Manager
Eastern Region & WNI Support

GGP:cam

cc: D. V. Shaller
[redacted]
J. G. Kern

DC-N-6015.1

Louis Caso

Response to Additional Information
on ECCS Analysis Requested by NRC — September 29, 1975

Donald C. Cook Nuclear Plant
Docket No. 50-315
50-316
DPR-58
CPPR No. 61

COPY

Mr. Karl Kniel, Chief
Light Water Reactors
Branch No. 2-2
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Kniel:

The following is in response to your letter dated July 10, 1975 requesting additional information in order to allow the staff to complete its evaluation of the Donald C. Cook Nuclear Plant, Unit No. 1 ECCS re-analysis. We consider this additional information only, as we believe our July 9, 1975 submittal of Amendment 64 to the Donald C. Cook Nuclear Plant Final Safety Analysis Report fulfills the requirements of the December 27, 1974 Order for Modification of License and meets the requirements of Appendix K to 10CFR50.

Item 1 - Break Spectrum and Partial Loop Operation

The provisions of the NRC memorandum entitled "Minimum Requirements for ECCS Break Spectrum Submittals" have been complied with as of our submittal of Amendment 64 to the Donald C. Cook Nuclear Plant FSAR.

Information on 3-loop operation is currently being evaluated. Until this review is completed, we will administratively restrict 3-loop operation in accordance with the requirements of the P-8 setpoint for four loop operation as provided in Table 3.3-1 on page 3/4 3-7 of the Technical Specifications of the Donald C. Cook Nuclear Plant.

Item 2 - Potential Boron Precipitation

We have supplied the information you have requested in our letter of April 14, 1975. Since that submittal, we have received additional information from Westinghouse regarding boron precipitation following a hot leg break and hot leg recirculation. This information was transmitted to the NRC in Westinghouse letter NOC-EB-364 dated July 23, 1975. A copy of this Westinghouse letter is attached to supplement our response of April 14, 1975.

Item 3 - Single Operator Error

The design of the Donald C. Cook Nuclear Plant, Unit No. 1 incorporates an analysis of possible failure modes, both active and passive, of ECCS equipment and their effects on ECCS performance. The ECCS system installed on the Donald C. Cook Nuclear Plant satisfies the General Design Criteria for Nuclear Power Plants (Appendix A of 10 CFR 50) as well as the provisions of Sections 50.46 and Appendix K to 10 CFR Part 50. The Westinghouse ECCS evaluation model (summarized within WCAP-8339) reviewed the range of possible failure modes of ECCS equipment and assumed the most damaging single failure of ECCS equipment.

The Donald C. Cook Nuclear Plant Design Criteria includes an analysis of valve failures that could adversely affect ECCS performance. As requested in your July 10, 1975 letter, we have made a further study of the ECCS valves and we have not been able to determine a failure mode of an electrical system component that could cause an undesired mechanical motion of a valve or other fluid system component. This is especially true on the Donald C. Cook Nuclear Plant - because all control circuits to these valves are provided with contacts on both the power side and the ground side of the control power, thus assuring that a single ground could not result in inadvertent valve actuation.

The present system of automatic confirmatory valve positioning signals following a LOCA, operating procedures, alarm and indication, and operator qualifications and periodic retraining precludes the likelihood of a mispositioned valve that could result in the loss of system safety function more limiting than that incorporated in the design criteria for the Donald C. Cook Nuclear Plant.

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Branch Technical Position EICSB 18 regarding single failure criteria of manually controlled electrically operated valves is inconsistent with past practice regarding the General Design Criteria for Nuclear Power Plants. It should be further noted that this type of single failure was not applied to the Donald C. Cook Nuclear Plant. In our judgment, there is no basis for applying this interpretation to the Appendix K ECCS evaluation model. Since this interpretation has not been made at the time the ECCS regulations were promulgated, it cannot be considered an appropriate interpretation of 10 CFR Part 50 Appendix K, Section I.D.1.

We firmly believe that Branch Technical Position EICSB 18 should not be applied to the Donald C. Cook Nuclear Plant for both the analytical reasons and the special design features described in the preceding paragraphs. We have however, defined the valves which may be affected by Branch Technical Position EICSB 18. These valves are listed below:

Valve Number	Function	FSAR Flow Diagram
IMO 910 IMO 911	Suction to charging pumps from RWST	6.2-1
IMO 261	Suction to RHR pumps from RWST	6.2.1A
IMO 390	Suction to Safety Injection Pumps from RWST	6.2-1
IRV 110 IRV 120 IRV 130 IRV 140	Accumulator drain valves to reactor coolant drain tank	6.2-A
IMO 51 IMO 52 IMO 53 IMO 54	Valves from charging pumps to reactor coolant system cold legs	6.2-1

It should be noted that valves IMO-910, 911, 261 and 390 will be required to change position following the postulated LOCA. IRV-110, 120, 130 and 140 are closed during the entire course of the accident, and IMO-51, 52, 53, and 54 are open during the entire course of the accident.

Item 4 - Submerged Valves

We have reviewed all the valves with motor operators inside of the containment with respect to ECCS performance analysis, and containment isolation, which included the requirements for limiting boric acid concentration during long term cooling.

The valve motor operators that will be subject following a loss of coolant accident include the foll

Valve Number	Function	FSAR
IMO-110 IMO-120 IMO-130 IMO-140	Accumulator Discharge Valves	6.2-1
IMO-51 IMO-52 IMO-53 IMO-54	Charging Pump Discharge to reactor coolant system cold legs	6.2-
IMO-128 IMO-129	Normal Cooldown from hot legs	6.2-1
IMO-111	Normal cooldown to cold legs	6.2-1
QMO-250	Reactor coolant pump seal water	6.2-1

It should be noted that each of the above valv is required to remain in a fixed position following i.e., valve operation of the above valves is not required following the accident.

Further review has shown that no spurious operation of the above valve motors will result from the submersion of the above valves. The details of this review are presented below:

These valves are AC motor-operated with power supplied at 600 V by starters located in motor control centers outside of the containment and, therefore, remote from the valve operator and the flood condition. The control for these valves is at 220 V AC and is supplied from control transformers dedicated to each valve starter. Valve operator heaters are powered at 120 V AC from separate valve heater distribution cabinets.

The control elements subject to flooding with the valve are the torque and position switches located in the valve operator. The power elements subject to flooding with the valve are the motor and heater.

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With a configuration such as used in the control and powering of the above valves, the only possible means of obtaining a spurious operation under the stated condition would be to energize the close or open starter coil in the valve control center. Bridging the torque or position switches with moisture, or even a perfect electrical conductor, will not result in energizing either of these coils because the control circuit is so designed that the control switch and/or safeguards relay in the control room isolate these coils from the remainder of the control circuit until such time as the operator or the safeguards logic elect to change valve position.

It might also be pointed out that flooding the above valves will not result in spurious operation of any other valves submerged or otherwise as there are no interlocks to other valves made on the subject valve position switches.

As a result of this review it is concluded that no design changes are required to solve potential flooding problems.

Item 5 - Containment Pressure

We are informed by Westinghouse Electric Corporation that the containment pressure used to evaluate the performance capability of the ECCS analysis is consistent with the Branch Technical Position CSB6-1, and that the model was approved by an NRC letter to Westinghouse dated May 30, 1975.

Item 6 - Low ECCS Relood Rate

The Westinghouse March 15, 1975 ECCS evaluation model was used in the Donald C. Cook Nuclear Plant Appendix K analysis, as noted in Appendix P to the FSAR.

Unit 2 - Information

The above analysis is applicable to the Donald C. Cook Nuclear Plant Unit 1 with 15 x 15 fuel. However, as we have informed you previously, it is our intent at this time to use 17 x 17 fuel. When the above information becomes available for operation with 17 x 17 fuel, it will

be supplied on a timely basis consistent with the initial Unit 2 core loading schedule.

Very truly yours,

John Tillinghast
Vice President

JT:ma
Enclosure

cc: G. Charnoff
R. Walsh
Robert J. Vollen
R. C. Callen
P. W. Steketee
R. W. Jurgensen - Bridgman
R. S. Hunter

bc: A. S. Grimes/R. L. Shoberg
S. J. Milioti/P. W. Daley
J. G. Feinstein
DC-N-6665.7
DC-N-6079
M.I. Olken/T.E. King/R.C. Carruth
DC-N-6015.1

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Question 40.6

State the extent of conformance of the safety systems to Regulatory Guides 1.41, 1.63, 1.75, 1.93. Also for each case where the D. C. Cook Unit Number 2 design does not conform to a particular recommendation contained in these Regulatory Guides, state the technical bases for the acceptability of the D. C. Cook Unit Number 2 design.

Response to Question 40.6

Regulatory Guide 1.41

The Electrical Design for D. C. Cook Plant complies with Regulatory Guide 1.41. Pre-operational verification tests demonstrating train independence will be performed. These tests will demonstrate the full redundancy of plant safety systems, their power supplies and control power requirements, AC and DC, in conformance with the regulatory position.

Regulatory Guide 1.93

D. C. Cook Technical Specifications are in compliance with Regulatory Guide 1.93.

Regulatory Guide 1.63

The electrical containment penetrations were purchased based on specifications developed in May, 1970. The electrical penetrations used at D. C. Cook Plant are in compliance with IEEE 317-1972 and Regulatory Guide 1.63 except as follows:

IEEE 317-1972

Section 4.1 Mechanical Design. The D. C. Cook penetrations were made in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for Class B vessels. Welds were magnetic particle tested in lieu of the radiographic method as per ASME Boiler and Pressure Vessel Code, Section III, Appendix IX, Article IX - 350.

Regulatory Guide 1.63

Paragraph C.1. The prototypes of the electrical penetrations were tested for the momentary values of the fault current. The overcurrent protective devices, either safety related or balance of plant, are all of the same quality level. Redundant protective devices are not employed.

Paragraph C.4. The penetrations were manufactured in accordance with Mil. Spec. Mil-Q-9858A which meets the functional requirements of ANSI N.45.2-1971.

Safety Guide 1.75.

The design of D. C. Cook Plant complies with the separation requirements of Safety Guide 1.75 as applied to Class IE equipment and circuitry. The design does not comply with the safety guide in the treatment of associated circuits.

Non Class IE cables are routed with Class IE cables in cable trays. The cable numbers of these associated circuits are modified to include a letter designation identifying the train association. These cables are allowed to leave the Class IE cable trays and be routed with non safety cables but are not allowed to be again routed with Class IE cables.

Power cable trays are of the open ventilated construction. Control and instrument trays are made of steel with solid sides and bottom. Tray covers are normally provided only where mechanical protection is required for the trays.

Cable tray installations where the minimum separation distances specified in 5.1.3 and 5.1.4, as applicable, are equalled or exceeded are not supplied with barriers. Where the specified minimum separation distances are not maintained, fire barriers consisting of 1/2" Maranite are installed between trays for horizontal separation or on the bottoms and sides of the upper trays for vertical separation. The minimum vertical separation distance where barriers are employed is one foot. The vertical arrangement of cable trays of opposite trains is permitted only where trays of one train cross trays of a different train. The barriers installed on the bottoms of the upper tray extended 18 inches beyond the crossing.

The maranite barriers attached to the bottoms and sides or mounted between trays prevents direct flame impingement on the trays and provides a fire proof barrier of extremely low thermal conductivity. Heat rising to the upper tray from a fire originating in a lower tray is directed around the upper tray and heat transfer to the upper tray is effectively reduced.

Question 40.10

Identify all electrical equipment, both safety and non-safety, that may become submerged as a result of a LOCA. For all such equipment that is not qualified for service in such an environment provide an analysis to determine the following:

1. The safety significance of the failure of this electrical equipment (e.g. spurious actuation or loss of actuation function) as a result of flooding.

2. The effects on Class IE electrical power sources serving this equipment as a result of such submergence, and

3. Any proposed design changes resulting from this analysis.

Response to Question 40.10

The lists below identify all electrical equipment that may become submerged as a result of a LOCA.

a. The motor operated valves which will be submerged as a consequence of a LOCA are the following:

IMO-110 " -120	Accumulator discharge valves
IMO-130 " -140	
IMO-51 " -52 " -53 " -54	Charging pump discharge to reactor coolant system cold legs
IMO-128 ICM-129	Normal cooldown from hot legs
ICM-111	Normal cooldown to cold legs
QCM-250	Reactor coolant pump seal water

These M. O. valves are the same as those listed in Mr. J. Tillinghast's letter to the NRC of September 29, 1975. These valves were studied and the comments made in Mr. Tillinghast's letter were re-verified. Further, the accumulator discharge valves power supply breakers, in the motor control center, are kept open to avoid the possibility of any inadvertent operation.

VMO-~~40~~ 80	Damper motor for hot sleeve ventilation fan
VMO-~~41~~ 81	" " " " " " "

After a LOCA the above two valves limit switches and motors will be flooded. The position of these dampers following a LOCA is not critical and their misoperation, if it occurred, will not affect the safety of the plant.

b. The solenoid valves which will be submerged after a LOCA are the following:

IRV-155	Test line from Accumulator tank #1 Valve
IRV-165	" " " " " #2 "
IRV-175	Test line from Accumulator tank #3 Valve
IRV-185	" " " " " #4 "
IRV-156	" " " RHR Injection Line 1 "
IRV-166	" " " " " " 2 "
IRV-176	" " " " " " 3 "
IRV-186	" " " " " " 4 "
IRV-50	Accumulator fill line valve from Boron Injection Tank
IRV-60	Accumulator fill line valve from Safety Injection Pumps
DRV-100	Reactor drain line to drain tank valve
DRV-150	Reactor coolant drain tank emergency drain valve
DRV-1	Pressure relief tank drain to reactor coolant drain tank valve
NRV-101	Hot leg loop #1 sample valve
QRV-150	RCP seal water bypass return valve
RRV-103	Pressure relief tank letdown to vent header valve

All these valves are operated from a 250 VDC source. The elements subject to flooding are the operating coil and limit switches. Study of their electrical control circuits showed that in case of a flood, shorting of the limit switches will not result in energization of the operating coil or changing in the position of the valve; this is so because control switches in the control room, that cannot be short circuited by the flood, effectively will isolate the operating coils from their power sources.

No interlock exists among these valves, or from them to any other equipment in the plant, therefore flooding of their limit switches and spurious coil energization, should it occur, would not result in a spurious operation of another device.

The above listed solenoid operated valves are closed and de-energized during normal operation; they all fail closed when de-energized. However, misoperation of any of these valves following a LOCA, should it occur, is of no significance for the safe shutdown of the plant.

c. The two lighting transformers inside the containment will also be flooded after a LOCA. These transformers are used for containment lighting and to provide power for the closed circuit TV in the containment. Operation of the closed circuit TV however is not required during a LOCA.

d. In case of a LOCA, electrical penetrations through the containment wall will also be below flood level. Electrical power and instrumentation penetrations were designed and built to be functional underwater. Of the electrical control penetrations, one per quadrant was designed and built to be functional under flooding conditions. In reviewing the electrical controls run through the non-waterproof electrical control penetrations, no devices were uncovered whose misoperation might jeopardize the safety of the plant. However, the circuit for the Hydrogen Recombiner remote temperature readout, though itself not essential, if short-circuited by the flood may burn the control transformer and jeopardize the operation of the hydrogen recombiner. The circuit to the remote temperature readout device was revised to prevent the possible failure of the hydrogen recombiner control power transformer.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR PROTECTION SYSTEM MODIFICATION

AMERICAN ELECTRIC POWER SERVICE CORPORATION

DONALD C. COOK, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

American Electric Power Service Corporation (AEPSC) has proposed to upgraded portions of the reactor protection system (RPS) instrumentation for the D. C. Cook, Units 1 and 2. This modification to the RPS is necessary due to the increased maintenance required on the existing Foxboro H-Line protection system and the difficulty in obtaining qualified replacement parts. The upgrade involves the installation of Foxboro SPEC 200 MICRO Line of micro-processor based instrumentation. This upgrade provides the licensee with configurable, micro-processor based, protection and control system modules. The SPEC 200 MICRO system processes the same inputs as the current analog system, performs the same calculation and bistable functions, and supplies contact outputs to the reactor protection logic for initiating a reactor trip and ESF functions. The system also incudes isolated analog outputs to indicators, recorders, plant computer and various control systems. The specific equipment and configuration of this upgrade has been previously approved by the NRC for use in the reactor protection system at Haddam Neck

in 1990. Therefore, the staff review of the D. C. Cook RPS Upgrade incorporated direct comparison to the previously approved Haddam Neck design, and focused on aspects specific to D.C. Cook.

2.0 OBJECTIVE

The objective of this safety evaluation report (SER) is to assess the adequacy of the proposed Foxboro SPEC 200 MICRO micro-processor based system, with emphasis on replacement system features that differ from the previously approved design. Attendant key issues addressed in the SER are the specific design bases for the new system, potential hardware vulnerabilities and susceptibilities, software development, verification and validation (V&V), configuration management, potential control/protection systems interactions, system failure modes and effects, reliability, and the developmental and operational history of the system equipment. The hardware and software design was reviewed against applicable NRC and industry standards including the requirements and guidance of IEEE Standard 279-1971 and ANSI/IEEE-ANS-7-4.3.2-1982.

3.0 TECHNICAL APPROACH TO THE EVALUATION

3.1 GENERAL

The licensee was requested to demonstrate compliance with the appropriate criteria and demonstrate that the use of micro-processor based equipment will not degrade the reliability of the RPS relative to the plant design basis. Pertinent criteria used in this review included the Standard Review Plan (SRP) and Regulatory Guide (RG) 1.152, "Criteria for Programmable Digital Computer System Software in Safety Related Systems of Nuclear Power Plants" which endorses ANSI/IEEE-ANS-7-4.3.2-1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Station." The staff reviewed the following major topics which are important to safety and required licensee/vendor support in providing appropriate documentation and personal interface with the reviewers:

1. Assessment of functional equivalence of the upgrade relative to the original design basis.

2. Assessment of the vendor's design (functional requirements & specifications) and verification and validation processes.

3. Assessment of licensee and vendor procedures and methodologies for the transformation of functional requirements to the software configuration that implements the requirements.

4. Assessment of the licensee's development of the design modification with respect to the design basis.

5. Assessment of the licensee's configuration management process for the new design.

These major topics are addressed from hardware, software, and systems engineering perspectives. In executing the review, a "thread concept" evaluation approach was used, whereby a single functional (sensor through actuator) requirement was followed from design through implementation, verification and validation testing, and any retesting required when a failure is encountered. Where necessary, additional functional requirements were examined to sufficiently cover the hardware, software, and systems evaluation criteria.

3.2 APPROACH TO HARDWARE AND SYSTEMS ASSESSMENT

The licensee was asked to demonstrate that hardware/system vulnerabilities and susceptibilities that have potentially greater significance in a digital system than in an analog system have been addressed in the design. It was requested that particular attention be placed on interfaces of the new digital equipment with old equipment or circuits including power sources. As part of the review, the licensee and vendor also provided evidence of the enhancements provided by the new equipment such as higher reliability and more accurate control capability, such as a reduction in instrument setpoint drift. Areas

of potential vulnerability/susceptibility addressed by the licensee and assessed in the staff review include the following:

1. Environment-temperature (particularly within cabinets) and humidity.

2. Electrical voltage and frequency surge withstand capability (SWC) credible faults (system tolerance to power surges), and the use of existing inverters to supply power to the new equipment.

3. Electromagnetic interference (EMI) including radio frequency interference (RFI) - radiated EMI/RFI susceptibly and noise propagation via power lines to the new equipment.

4. Failure Modes and Effects -
   a. Detectability of failures and recovery.
   b. Special failure modes (for example, system stall, timing errors/instability) and the effects on frequency and severity of transients.

5. Technology Upgrade Comparisons -
   a. Comparison of reliability/availability of new equipment to of the original equipment.
   b. Identification of failure mode differences between new and old equipment.

6. Independence -

   a. Physical and electrical separation and barriers.

   b. Isolation devices and their application.

   c. Independence of various software functions.

7. Seismic Qualification.

To effectively perform the review, the licensee was requested to provide sufficient description of the design change to enable a comparison of the original RPS to the proposed RPS upgrade. The licensee provided the following documentation:

1. A functional block diagram of the RPS.

2. A hardware block diagram of the RPS including interfaces (before and after the upgrade).

3. Clear definition on the above drawings of protection/control system boundaries protection division (channel and train) boundaries power sources/supplies and isolation devices or other methods of assuring independence that are used at these interfaces.

4. The modification package, supporting design documentation, and safety evaluation.

5. Identification of the standards, criteria, specifications, calculations, analyses and tests that address the hardware attributes of interest. The criteria, assumptions, and methodology used to establish the design basis for the electrical environment and EMI environment, and demonstration of compliance to the design basis were of special interest.

Also, the licensee provided or made available for review the following:

1. Factory acceptance test reports

2. Site acceptance test reports

3. Operating and maintenance procedures

4. Test procedures/reports for periodic testing of the system

5. Data on self-diagnostic failures

3.3 APPROACH TO SOFTWARE ASSESSMENT

The inclusion of microprocessors and their concomitant software in the RPS marks a significant departure from the original analog electronic design. While the transition to digital systems may provide significant performance and safety advantages, it may also introduce issues and concerns that have not

been encountered previously and have not undergone a thorough safety review at D.C Cook.

For the software assessment, the licensee must demonstrate that the software is in accordance with the functional design, and that the implementation process will result in reliable software. The licensee must also demonstrate that an adequate program is in place to assure that the reliability of the software is maintained for the entire software life cycle including revisions to the permanent software and user configurable software.

To assess these points of concern, the software audit focused on three main areas:

1. The design, development and operational history of the vendor's software.

2. The procedures and methodologies in use by the licensee/vendor to ensure that the vendor supplied hardware and software control blocks are functionally equivalent to the design basis for the RPS.

3. The procedures and V&V processes used by the licensee/vendor to control the user configurable software elements.

To perform the software assessment, the licensee/vendor was asked to make available the following information:

1. A description of the design development and operational history of the vendor's software components.

   a. A description of the design and software development activities.

   b. The scope and results of static and dynamic tests for the software.

   c. A description of the acceptance testing performed and the results of the testing.

   d. Documentation of all software modifications performed since its release.

   e. A description of the PROM "burn in" process and what procedures are in place to ensure that the finished PROM is correct.

2. A description of the procedures and methodology used by the licensee to ensure that the functional design basis is implemented.

   a. The functional specification for the system and the derivation of the control block configuration.

b. The procedures and methods used to ensure that the control block configuration will implement the control logic as specified.

c. A description of the licensee's user configurable software management program.

d. The results of testing the completed system, the list of errors, and how the errors were detected and corrected.

3. A description of the procedures and V&V processes used to control the configuration of software for the control blocks.

a. A description of the vendor's verification activities for the control block software.

b. The results of verification activities during the development including characterization of the errors found by type and percentage of the total errors.

c. The results of validation activities during testing including characterization of the errors found by type and percentage of total errors.

d. A description of the steps taken to ensure the independence of the verifier and the verifiers recourse in the event that discrepancies or errors are found.

4.0 DESCRIPTION OF SYSTEM UNDER REVIEW

4.1 SCOPE OF SYSTEM MODIFICATION

The proposed RPS modification is limited to replacing power supplies and process signal conditioning instrumentation. The process signal conditioning instrumentation prepares signals from existing process sensors, determines when protective action is required, and initiates a channel trip at a predetermined limit. The modification does not change the existing RPS actuation logic. The licensee chose a design approach that limits duplication of existing functions of the RPS instrument cabinets and minimizes the impact to external cabling and power sources.

4.2 FUNCTIONAL/SYSTEM DESCRIPTION

The proposed changes are intended to modernize the D. C. Cook Reactor Control and Protection System instrumentation by replacing a portion of the original equipment supplied for processing the following parameter inputs: Pressurizer Pressure, Pressurizer Level, Reactor Coolant Flow, SG Narrow Range Level, SG Steam Flow, SG Pressure, Feedwater Flow, RCS Delta Temperature, RCS Average Temperature, Containment Pressure, Turbine Impulse Pressure, Wide Range RCS Temperature, Wide Range RCS Pressure, RWST Level, Containment Level, Auxiliary Feedwater Flow, TSAT Input Select, and Condensate Storage Tank Level. To implement the input processing, the licensee has procured a series of microprocessor-based control modules that have been installed in the existing

Foxboro Spec 200 racks at the D.C Cook Nuclear Plant. The following new equipment will perform the same functions as the equipment it replaces:

1. Foxboro SPEC 200 Type Analog Input Signal Conditioning, which changes the various types and values of input signals from the sensors into a common type of analog output signal that represents the input values at the channel level.

2. Foxboro SPEC 200 MICRO Digital Signal Processing Equipment, which takes the analog signal from the SPEC 200 analog input equipment at the channel level and:

   a. Changes the analog signals to digital signals.

   b. Processes the digital signals and compares them against predetermined limits, as well as performs dynamic functions and calculations.

   c. Changes the processed digital signals back to analog output signals.

3. Foxboro SPEC 200 Analog Output Signal Conditioning Equipment, which takes the analog output signals from the SPEC 200 MICRO and conditions them for use in the control system, as well as for indication and recording use.

4. Foxboro SPEC 200 Type Contact Output Equipment, which produces discrete trip signals from the SPEC 200 MICRO and provides trip signals for input to the reactor protection logic equipment.

5. Foxboro SPEC 200 Type Power Distribution Equipment, which powers the signal conditioning and processing equipment discussed above.

6. 75 VDC Multi-loop and + 15 VDC Multi-nest Power Supplies, which provide power for the field transmitters and SPEC 200 equipment.

4.3 HARDWARE DESCRIPTION

The new hardware is based upon the application of the Foxboro SPEC 200 MICRO Control System. The system consists of control cards mounted within the existing cabinets and display stations located at the various control panels within the control room. The control card is a rack-mounted microprocessor based unit which performs signal conditioning, regulatory control and logic control functions. The control functions are provided by the user configuration of up to 6 control blocks from a menu of 21 different types. Most block types have multi-functions available. The Continuous Display Station provides user interface to logic control blocks. These displays provide sequential paging to display more than one control block. Two nine digit alphanumeric readouts provide configurable loop tag identification, digital readout of the block parameters in percent or engineering units, and the ability to modify control block tuning parameters.

The hardware items were tested to and accepted against the requirements of IEEE 344-1975 and portions of IEEE 323-1974. Environmental specifications to

provide compatibility with the control room environment were identified for the equipment. The units are powered from the vital buses.

4.4 SOFTWARE DESCRIPTION

The SPEC 200 MICRO is a software based control system that uses a digital card to implement a wide range of control structures and logic. The SPEC 200 MICRO software was based on general functional requirements derived from the collective functions of approximately 100 existing Foxboro SPEC 200 analog control cards. The software is presented as 21 modules (control blocks) that can be configured for specific applications by a menu driven set up process. The responses selected from the menu by the licensee fill out internal tables which then control the operation of the specific control functions and logic.

On a single digital control card, an executable module for each of the 21 Processor Control Block is contained in the programmable read only memory (PROM) and cannot be changed by the licensee. Up to 6 of these modules can be configured by the licensee at the site on any one control card to implement the required control logic. The system is configured by using a personal computer. The configuration software provides menu driven "fill-in-the-form" displays. These displays allow the control scheme to be configured, or the control blocks contained within the control card to be modified. Configuration is the process of selecting control block types and interconnecting these blocks to produce the desired control scheme.

Displays allow the monitoring of the system parameters to verify selected control schemes.

The licensee stated that the D. C. Cook RPS application uses only six of the available 21 Control Blocks: CALC, ALRM, LLAG, GATE, SSEL, and RAMP. A brief description of these blocks follows:

CALC provides facilities for executing simple arithmetic equations. Fourteen functions are available and can be combined with data inputs and intermediary results to form an equation. The format is similar to that used on programmable calculators.

ALRM provides the facilities for setting alarm points for data inputs from the sensors. It includes absolute alarms, deviation alarms, rate alarms, output alarms and output limiting.

LLAG provides the facilities for setting lead-lag compensation parameters. The lead is defined by the configurator as an amplification factor of the lag term, and the lag is defined as a time constant.

GATE provides eight, two-input logic gates which allow configuration of various Boolean algebra (and, or, etc.) functions.

SSEL provides the facilities for selecting from up to eight separate input signals including the highest, lowest, or median.

RAMP provides a dual linear ramp generator with a single output.

5.0 TECHNICAL EVALUATION

The scope of the staff review of the Foxboro upgrade of the D. C. Cook Nuclear Plant, Unit 1 and 2 Reactor Protection System (RPS) includes the new RPS equipment to be installed, the interfaces of that equipment with existing equipment, and the environment encompassing the new Foxboro RPS equipment. This evaluation is based on the guidelines of R.G. 1.152. R.G. 1.152 states that the NRC staff encourages the application of advanced technology such as programmable digital computers in the operation of nuclear power plants if such technology serves to enhance safety. R.G. 1.152 endorses, with certain exceptions, IEEE Standard 603-1980 which establishes criteria for safety systems, structures and equipment necessary for plant safety.

Section 5 of IEEE 603-1980 establishes the criteria for safety system performance and equipment. The following sections discuss these criteria and the implementation of these criteria as established during the audit of the Foxboro facility and documentation. Included in the scope of the staff review against this criteria was the licensee's channel setpoint methodology, human factors considerations, grounding, electromagnetic and radio frequency interference, use of the configurator, and various tests that establish the acceptability of the Foxboro SPEC 200 and SPEC 200 MICRO equipment installed for this application.

Sections 6 and 7 of IEEE Standard 603-1980 establish criteria for sensing, command, execute, and functional design requirements. The staff review addressed these items as discussed subsequently.

Section 8 of IEEE Standard 603-1980 addresses power sources. The power for the upgraded RPS did not require change. The existing power sources were found by the licensee to be capable of supplying the additional loading associated with the upgrade. Therefore, the power sources are acceptable.

5.1 COMPLETION OF PROTECTIVE ACTION

Section 5.2 of IEEE Standard 603-1980 addresses the completion of the protective action by a safety system. If started either manually or automatically, the intended sequence of protective actions is to carry through to completion. Deliberate operator action is required to return the safety systems to normal. Seal-in of individual channels is not required. The upgraded RPS equipment consists of modules that affect individual channels. Individual channel trips may occur with no reactor trip. These individual channel trips are automatically removed if the process (level, pressure, temperature, etc.) returns to the non-trip side of the system setpoint. The upgraded equipment does not affect the Solid State Protection System where the individual channel trips are combined and the seal-in process occurs. Two or more channels must be in a trip state concurrently to affect a safety system actuation. The replacement equipment is only in individual signal processing

channels and, as such, supports the completion of a safety function. This is acceptable to the staff.

5.2 QUALITY

Section 5.3 of IEEE Standard 603-1980 states that the quality of the components and modules is to be consistent with minimum maintenance requirements and low failure rates. Safety system equipment is designed, manufactured, inspected, installed, tested, operated, and maintained in accordance with a prescribed quality assurance program.

5.2.1 Hardware

Section 6.0 of the American Electric Power Service Corporation specification for the subject equipment addresses quality, documentation, testing, and checkout requirements. The quality requirements include:

1. IEEE Standard 323-1974 as supplemented by IEEE Standard 381-1977

2. IEEE Standard 344-1975, and the requirement to produce and furnish the equipment "under an approved QA program that complies with 10 CFR 50, Appendix B"

The Foxboro Company Quality Assurance Program is governed by Foxboro document

CQA-2. This document was reviewed by the staff during the audit inspection and found acceptable for the upgraded RPS components. It is utilized throughout various Engineering Operating Procedures. Based on the Foxboro Company Quality Assurance Program and the above specification quality requirements, the hardware satisfies the quality requirements of IEEE Standard 603-1980.

5.2.2 Software

R.G. 1.152 "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," specifies that ANSI/IEEE-ANS-7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," is the basis for the staff review of digital systems after 1985. Section 4.1 of ANSI/IEEE-ANS-7-4.3.2-1982 addresses the software development plan. Selected portions of IEEE Standard 730.1-1989, "IEEE Standard for Software Quality Assurance Plans," have been used by the licensee to describe and document its compliance to Section 4.1 of ANSI/IEEE-ANS-7-4.3.2-1982. The software portion of this upgrade can be divided into two parts: (1) Executive and Control Block algorithms running on the control cards and (2) data used to configure the specific functioning of the Executive and Control Block algorithms which is developed and maintained by the hardware vendor, packaged as firmware (PROM), and can not be modified by the licensee. The licensee may, however, setup and modify configuration files that control how the firmware operates. The process of assuring the quality of the software, therefore, involves looking

at 1) the vendor's software life cycle for the Executive and Control Block firmware 2) the mechanisms utilized by the licensee for establishing configuration requirements, and 3) the means for verifying and maintaining change control of that configuration. The quality of the configuration management process used to create and maintain the data fed from the Configurator to the control cards is addressed in Section 5.11 of this report. The remainder of this section discusses the quality of the vendor-supplied firmware.

The vendor's software development process was previously audited by the NRC for the Haddam Neck RPS upgrade. Several aspects of the process were noted at that time including lack of independence during reviews and inadequate documentation of the organizational structure for review of test results, and follow up actions taken as the result of the reviews. These concerns were subsequently resolved as discussed below:

Methods used to assess software quality included 1) review of the licensee-provided Qualification Compliance Report (Vendor's Standard CES 281) which describes how each of the applicable portions of ANSI/IEEE-ANS-7-4.3.2-1982 and IEEE Standard 730.1-1989 were met, 2) review of the documents referenced in this report, and 3) follow-up interviews with vendor personnel, as needed. Information provided in the Qualification Compliance report served as a guide to the documents and processes reviewed.

According to ANSI/IEEE-ANS-7-4.3.2-1982, Section 3.2, the requirements

for safety systems utilizing software shall be documented and verified. Some 12 areas are defined that require specification. The vendor's functional specification for the SPEC 200 MICRO CCA Control Card Software along with the design documentation was examined. As augmented by various other formal documents, all areas requiring specification were properly documented. The sign-off and review history for these specifications was also examined and found acceptable.

According to ANSI/IEEE-ANS-7-4.3.2-1982, Section 3.3, the requirements for the integration testing shall be documented and verified. Hardware and software integration testing, in the case of this upgrade, refers to both the vendor testing done during the development of the control card hardware and software, and the testing of the configured system. The test plan for the software was developed prior to release by the vendor and executed at that time, but the results of the test were not formally documented. To rectify this, the vendor re-ran the test procedures as documented in the Foxboro Report 92-9024, "Evaluation of SPEC 200 MICRO Control Card and SPEC 200 MICRO Displays," May 1992. This factory acceptance test has been designated by the licensee as the hardware and software integration test procedure for acceptance of the software and hardware.

ANSI/IEEE-ANS-7-4.3.2-1982, Section 4.0 specifies that the software development process shall consist of several phases including, as a minimum, a development plan phase, a design phase, and an implementation phase. Verification of each phase is required. Compliance to the general requirement relating to the development plan has been demonstrated through compliance to

IEEE Standard 730.1-1989. Section 3.3 of IEEE Standard 730.1-1989 requires that the project management be described in terms of organization, tasks, and responsibilities. This information was reported in the Qualification Compliance Report. The vendor resolved the independence issue that surfaced during the Haddam Neck review by redoing the questionable walkthroughs and tests. In all cases, the repeat reviews included individuals outside the program development team.

Section 3.4 of IEEE Standard 730.1-1989 specifies that the documentation governing the development, verification and validation, use, and maintenance of the software be identified, as well as the methods for checking the accuracy of that documentation. Included in the list of minimum documents to be identified are software requirement specifications, software design descriptions, software verification and validation plans, software verification and validation reports, and user documentation. Details relating to each of these types of documentation are adequately provided in the Quality Compliance Report as well as in Foxboro Report QOAAE03. Appendix C of QAOAAE03 includes a listing of documents, walk-through dates, reader, and recorder. It should be noted that the types of documentation that were produced and maintained for the SPEC 200 MICRO do not match exactly with the names called out for the documents in the standard, however, the staff confirmed that their purpose and function as described in the Qualification Compliance Report was equivalent to the criteria of the standard. Of particular note was the lack of a software verification and validation plan during the original software development. As explained in QOAAE03, the development team did follow an organized methodology to verify and validate

the software developed. A recently released report of the software verification and validation was provided, as noted in Foxboro Engineering Document EDO2134, Revision C.

Section 3.5 of IEEE Standard 730.1-1989 provides criteria for software design as detailed in the Quality Compliance Report, the vendor's corporate engineering standards were in place from the inception of the project, including Corporate Engineering Standards (CES) 281:18, 19, 22, and 24 and CES 280:1. The Qualification Compliance Report also describes how well the vendor complied with these standards, practices, and conventions. During the staff review, an entire software function (thread) was traced for the GATE, SSEL, and RAMP algorithms to verify that the above CES actually existed and were followed during the software design. CALC, ALRM, and LLAG algorithms were previously reviewed by the staff during the Haddam Neck review. In the case of the standards regarding vendor technical reviews, there appeared to be less than adequate compliance, especially in the completion of review documentation and correspondence. However, the reviews themselves appeared to be properly conducted, as ascertained by direct interviews with one of the reviewers and the supervising managers. Several of the vendor's quality assurance standards have been upgraded recently, including CQA 3.3.1 to require more detailed documentation. The vendor, in response to the previous audit findings of inadequate documentation and independence, re-performed a number of the software walkthroughs to confirm its adequacy.

Section 3.8 of IEEE Standard 730.1-1989 provides guidance on documentation of software problems. The staff verified compliance with this criteria through

interviews with the vendor's staff and review of the problem report and release list included in the SPEC 200 MICRO Panel System Test and Evaluation Report. The latest problem report was dated May 10, 1987. This problem was corrected in release version SA-11 which was released in May 1987.

Compliance with the criteria of Section 3.9 of IEEE Standard 730.1-1989 concerning software verification and validation (V&V) is detailed in the Qualification Compliance Report. Several reports produced through the use of the V&V tools discussed in the report including the Duane Growth Concept (Foxboro Report Number 87-SRR-001F, "Control Card Software," February 1987) and the McCabe Cyclometric Complexity Index (Foxboro Report Number 87-SRR-008F, "Software Complexity Analysis") were reviewed as was the overall description of the methods for performing white box testing (Foxboro Report QOAAE03 Revision B, "SPEC 200 MICRO Software Validation and Validation," October 1988).

Sections 3.10, 3.11, and 3.13 of IEEE Standard 730.1-1989 which contain criteria on the software change process are governed by Foxboro standards CES 281:15, CES 281.11, and CES 280:20, respectively. The general change and document control process used by the vendor was also discussed with vendor representatives. No problems were noted and the vendors procedures were properly exercised.

As described above, the staff verified the existence of the referenced software documentation and compliance to standards through a systematic trace (thread audit) of the documentation relating to the GATE, SSEL, and RAMP

control algorithms. Problems were found with the accuracy and completeness of some of the record-keeping in terms of dates, times, and findings of various walkthroughs. However, when problems were noted during the reviews, there appeared to be a complete documentation trail to prove that those problems had been adequately addressed and the necessary changes incorporated in the specification, design, or code as appropriate.

At a higher level, the staff assessed the corporate attitudes and climate relative to the importance of ensuring software quality. Interviews were conducted with various project members, project leaders, and quality representatives. These personnel were asked to discuss their views and attitudes regarding software quality. Without exception, those interviewed demonstrated a good awareness and an internalized commitment to corporate quality standards and procedures and the importance of that commitment. Some personnel were very aware of national quality trends relative to software engineering and are working to incorporate the latest concepts and approaches into their work. The staff was made aware of several upgraded vendor quality standards and found no evidence of forces or trends that would undermine the future quality of the software. The staff finds that acceptable quality assurance is provided for the RPS upgrade.

5.3 EQUIPMENT QUALIFICATION

Section 5.4 of IEEE Standard 603-1980 provides criteria or equipment qualification. Qualification of Class IE equipment is accomplished by meeting the criteria of IEEE Standard 323-1974. The qualification of safety system

equipment is established by type test, previous operating experience, analysis, or a combination of these methods. Qualification ensures the capability for system hardware to meet the performance requirements specified in the design basis.

The RPS equipment is located in the control room which is a mild environment. The purchase specification lists the qualification temperature range as 0°C to 50°C at ambient pressure. The humidity ranges between 10 percent and 90 percent. The maximum post-accident dose of radiation is specified at 1E+4 Rads, total integrated dose.

The staff identified a discrepancy in that the equipment manufactured by Foxboro is rated for operation down to 5°C rather than the 0°C specified lower limit. Per the Updated Final Safety Analysis Report (UFSAR), the control room normal ambient environment is maintained at 75°F±15°F. Heat rise occurs within the RPS instrument racks due to their operation. Because of the controlled environment and the rack heat rise, it is unlikely that the ambient temperature at the RPS equipment will drop below 5°C. Therefore, the lower operating limit of 5°C is acceptable.

Heat rise testing within cabinets similar to the D. C. Cook AMCO instrument racks was documented in Hurst Engineering report 2985-HEI-05. That testing showed a maximum heat rise of 9°C within the AMCO rack. That, when added to the maximum control room ambient temperature of 50°C specified in the UFSAR is within the qualification temperature limit tested for the Foxboro equipment (60°C). Therefore, the Foxboro equipment is suitable for operation within the

D. C. Cook control room temperature limits. Additional margin is realized since the normal ambient upper limit of the control room temperature, per the UFSAR, is 90°F.

Seismic qualification was established by type testing of the Foxboro components and analysis of the Foxboro equipment in the existing racks. The qualification testing encompassed both the operating basis earthquake and the safe shutdown earthquake acceleration values for the D. C. Cook Nuclear Plant site. The following standards and criteria were used to establish satisfactory seismic qualification: IEEE Standard 344-1975; IEEE Standard 381-1977; NRC Reg Guide 1.100; EE Standard 420-1982.

The seismic qualification testing and analysis is documented in Foxboro Type Test Report QOAAB01 and Hurst Engineering Reports 2985-HEI-07 and 2985-HEI-08. The materials inspected during the audit document the capability of the Foxboro SPEC 200 MICRO components installed in the AMCO instrument racks to remain functional before, during, and after an operating basis and safe shutdown earthquake at the D. C. Cook Nuclear Plant.

Voltage surges in the digital RPS can occur as the result of lightning, switching transients, electrical faults, and other fast transients. Testing for surge withstand capability was specified by the licensee to meet the criteria of IEEE Standard 472-1974. However, testing per Foxboro Test Plan IT-220, which was completed in June 1993, follows the criteria of more recent standards such as ANSI/IEEE Standards C62.41-1991 and C62.45-1987. The testing covers electrostatic discharge, common mode rejection, normal mode

rejection, high frequency transients, surge withstand capability, and lightning effects. The staff confirmed that the proposed testing methodology is acceptable to demonstrate surge withstand capability. The current injection method of testing for electrostatic discharge, ranging between 2 kVdc and 6 kVdc, was applied to operator accessible surfaces. The staff reviewed the surge withstand capability test results documented in IT-22A, Rev. 0, dated July 16, 1993, and found the results acceptable.

Surge withstand capability will be tested at 2.5 kVdc for common mode surges and at 1.5 kVdc for normal mode surges, at a burst frequency of 120 Hz for two seconds. Lightning effects will be simulated by 0.5 kVdc and 1.0 kVdc in the normal mode and 1.0 kVdc and 2.0 kVdc in the common mode on the system. Electromagnetic and radio-frequency interference withstand capability is discussed in Section 5.15 of this report.

The replacement of wiring and cables is minimized by the design of the RPS upgrade project. Replacement equipment is to be mounted in existing instrument racks. The licensee committed that any replacement cables with splices and connections will comply with the criteria of IEEE Standard 383-1974.

The staff finds that the upgrade of the D. C. Cook RPS with Foxboro SPEC 200 modules has been demonstrated to meet applicable equipment qualification requirements, and the system is, therefore, capable of acceptable operation in the anticipated environmental conditions.

5.4 SYSTEM INTEGRITY

Section 5.5 of IEEE Standard 603-1980 specifies that a safety system be designed such that it accomplishes the safety function under the full range of applicable design basis conditions (system function). The RPS upgrade to install Foxboro equipment updates the components of each of the four channels and does not effect the original system functional design or logic. The system integrity is maintained with this upgrade of system components. Therefore, the requirements of Section 5.5 of IEEE Standard 603-1980 are satisfactorily met.

5.5 INDEPENDENCE AND PHYSICAL SEPARATION

Section 5.5 of IEEE Standard 603-1980 also specifies a design where 1) there is independence and physical separation between redundant portions of a safety system, 2) there is independence and physical separation between the safety system equipment and the effects of a design basis event, 3) there is independence and physical separation between safety systems and other systems; and 4) a single failure in a non-safety system will not cause a design basis event which negates the safety system function designed to mitigate that event. There are specific requirements on the classification of equipment, isolation between safety system equipment and non-safety system equipment, and separation from (barriers between) safety system equipment and non-safety equipment.

The D. C. Cook Plant has a drawing control program that controls the physical equipment layout, design analysis and evaluations, adjustments, calculations, and system configuration. Foxboro designed the RPS replacement components into the existing system consistent with the plant drawing control program. The Foxboro design responsibilities included functional, rack loading, wiring, termination, and power supply drawings, along with database configuration information. The database configuration information includes the Configurator settings for the SPEC 200 MICRO modules as discussed in Section 5.11 of this report.

To confirm acceptable provisions for system isolation. the staff examined a N-2AO-V2H voltage to current (V/I) isolator. This module is a custom factory modification of a standard V/I converter. Power supply isolation is added to the circuit board. This allows the module to withstand a maximum credible fault, as documented in Foxboro Report 92-0029a. Testing of the RPS modification included the application of 1) grounded output leads, 2) 250 Vac rms at 60 Hz from both output leads to ground, 3) 250 Vac rms at 60 Hz across the output leads, 4) 480 Vac rms at 60 Hz from both output leads to ground, and 5) 480 Vac rms at 60 Hz across the output leads. System isolation is provided for the safety signal to non-safety uses such as control room displays. The testing satisfactorily demonstrated that the module meets the isolation criteria of Section 5.6.3.1 of IEEE Standard 603-1980.

The Foxboro equipment is being installed in the existing RPS instrument racks. The Configurator is physically remote from the RPS cabinets and is not connected to any module in the RPS racks. The staff finds that acceptable

independence and separation are, therefore, maintained per the criteria of IEEE Standard 603-1980.

5.6 CAPABILITY FOR TEST AND CALIBRATION

Sections 5.7 and 6.5 of IEEE Standard 603-1980 contain specific criteria regarding safety system equipment testing and calibration. Testing of Class IE Systems should be accomplished in accordance with the guidance of ANSI/IEEE Standard 338-1977.

The capability for periodic testing and calibration is retained in the Foxboro SPEC 200 MICRO instrument upgrade. The test panel supplied with new equipment will provide comparable capability for test and calibration utilizing present manual methods and permanently installed test facilities very similar to what has been in use with the existing H-Line equipment. In fact, additional test points are provided with the new system to enhance in-place testing and calibration of rack mounted components.

The SPEC 200 MICRO control cards are individually removed from the instrument rack for configuration and calibration. The control card is configured using a personal computer (PC) with Foxboro configuration software (Configurator). This software provides menu-driven "fill-in-the-form" monitor displays for entry of initial settings. The initial settings (configuration) for each channel, including ranges, setpoints, deadband, and lead/lag functions, are downloaded to the control cards as required for each function using the menu-

driven displays and the PC keyboard. Once the configuration is complete, the Configurator is disconnected from the control card. The control card is then reinstalled in the instrument rack for service. An on-board lithium battery maintains the installed configuration while the control card is in transit.

Channel Functional Testing will be accomplished utilizing present manual methods and installed test facilities due to the similarity in channel architecture between the original and modified system. The licensee plans to use existing H-Line equipment procedures with minor changes to perform Channel Functional Testing.

By letter dated December 22, 1993, the licensee responded to a concern regarding overlap testing. The licensee state that the first rack test point is the key overlap point and the loop current is monitored at each calibration test segment. The test program results in a total end-to-end loop check through overlapping. The licensee stated that this test methodology is the past and current D. C. Cook plant practice and is common for the industry. The licensee further stated that the test point resistor tolerances do not impact loop accuracy or performance. There are no effects caused by test point resistor tolerances because these effects are calibrated out as a result of the system design and the D. C. Cook plant calibration methods. Test point resistor tolerances are invisible to the operators and to the performance of automatic actuation functions.

The staff finds that the test and calibration capability and procedures are acceptable, and in conformance with the criteria of IEEE 338-1977.

5.7 INFORMATION DISPLAY

Section 5.8 of IEEE Standard 603-1980 addresses the criteria for safety system information displays. These criteria encompass information displays for system status indication, bypasses, and manually controlled actions. Displays for manually controlled actions must also be in accordance with the guidelines of R.G. 1.97 for post-accident monitoring instrumentation.

The proposed RPS upgrade contains no new information channel displays, and the existing displays were not disturbed. The existing information displays included in the D. C. Cook RPS design have been previously reviewed and approved by the staff.

5.8 CONTROL OF ACCESS

Section 5.9 of IEEE Standard 603-1980 addresses the criteria for administrative control of access to safety equipment. Administrative control of access to the SPEC 200 MICRO equipment is provided by physical access control of the installed equipment, to configuration hardware, software, and data. The RPS cabinets are kept locked and the keys are under administrative control. Configuration hardware, software and data are controlled via plant procedures and policy.

The administrative control of access to the Foxboro hardware, software, and data was reviewed by the staff. The staff determined that all hardware, software, and configuration changes are adequately controlled through approved procedures. The procedures include reviews and sign-offs to ensure that the impact of all changes are properly implemented throughout all plant operating aspects (e.g., procedures, training, drawings, maintenance, and testing).

The Foxboro RPS modules are located in existing instrument racks that have key-locked doors. The existing equipment in those racks will be removed, except for the terminal blocks and the field wiring connected to those terminal blocks. Any rack door that is open causes an annunciator to light in the control room (one annunciator module per RPS channel). The control room itself has restricted access. Work done on the RPS is controlled and administered by procedure and key control. While there are zero and span adjustments on the Foxboro SPEC 200 analog modules, access to those modules is controlled. The Configurator is physically remote from the RPS cabinets and is not connected to any module in the RPS racks. The staff finds that acceptable access control to the RPS is provided.

5.9 IDENTIFICATION

Section 5.11 of IEEE Standard 603-1980 addresses the criteria for the identification of safety system components and associated documentation. The distinct identification for each redundant portion of a safety system is to be in accordance with the guidelines of IEEE Standard 384-1977 and

IEEE Standard 420-1973. In addition, the associated documentation is to be distinctly identified in accordance with the guidelines of IEEE Standard 494-1974.

The licensee has committed to retain channel identification consistent with the channel identification of the original H-Line equipment. Redundant channel cabling is color coded, and the RPS instrument cabinets are appropriately labeled in accordance with the above IEEE standards. New identification labels will be installed at the nest and rack level for all new equipment. The licensee also stated that components with dedicated functions, not interchangeable with other components will be distinctly labeled to ensure configuration at the component level. Drawings, procedures and references are to be updated to reflect new equipment at the rack, nest, and component level upon installation.

The new RPS cabinets were inspected by the staff in the plant shipping/receiving area prior to equipment installation. The plant identification labels and color coding on the cabinets and components were determined to be acceptable. The labels are clearly legible, and consistent with plant documentation and drawings. Color coding was implemented on a channel level as well as train level, which is consistent with the original plant design, and is, therefore, acceptable to the staff.

5.10 AUXILIARY SUPPORTING FEATURES

Section 5.12 of IEEE Standard 603-1980 has specific criteria regarding the use of auxiliary supporting features. IEEE Standard 603-1980 defines an auxiliary supporting feature as systems or components which provide services (such as cooling, lubrication, and energy supply) that are required for the safety systems to accomplish their safety functions. The RPS instrument racks at D. C. Cook have open mesh screening at the top of the cabinets and louvers in the rear door. These features support natural circulation of the control room air for component cooling by convection. The Foxboro RPS equipment used at D. C. Cook does not require forced circulation (cooling fans) or lubrication.

5.10.1 Lithium Batteries

One lithium battery is used in each SPEC 200 MICRO module. The sole function of each battery is to retain the random access memory (RAM) configuration of the associated microprocessor when the module is without normal power. The primary battery function is to retain the software configuration when the module is in transit from the RPS instrument racks to the Instrument Calibration Lab for configuration or maintenance and in transit back to the instrument racks. The battery also retains the software configuration when there is a momentary power loss, such as the few milliseconds involved during the transfer of power between the normal inverter-supplied power and the backup ac power source. Documentation provided by the licensee states the CMOS chip memory is guaranteed at supply voltages

of 2 Vdc or higher.

Lithium batteries have an inherent long life in standby applications due to the chemical composition of the battery. The lithium electrolyte construction is ideal for memory backup power applications such as this where the battery is essentially unloaded for more than 99 percent of its life. In particular, the licensee stated that the three volt lithium battery cell used in this application is able to provide required memory retention for 15,000 hours if the cell is new. Memory is retained for a minimum of 1,000 hours when the cell reaches the low voltage threshold. This threshold is continuously monitored by the SPEC 200 MICRO module. A light emitting diode (LED) is illuminated when the battery has less voltage than the threshold being monitored. It was noted that the SPEC 200 MICRO modules are shipped with the batteries removed, and the batteries are shipped separately.

As each battery is installed, its serial number, lot number, and installation date are included in a database. Surveillance procedures call for monitoring the battery status, and will direct replacement of the battery before the useful battery life expires. Additionally, routine preventive maintenance will replace the batteries before the end of useful battery life as necessary. Through the use of the database, trends in battery performance can be observed. The staff finds the procedures associated with the lithium batteries to be acceptable for maintenance and surveillance.

The batteries were tested using Foxboro test procedure BT-180A. The test demonstrated acceptable battery characteristics when the battery terminals

were shorted or when the battery was installed in reverse polarity. The fresh battery had the same potential when loaded or unloaded. The SPEC 200 MICRO module performed as though the battery was not present when it was installed with reverse polarity. With the battery terminals short circuited, the battery temperature rose to 262°F after 7 minutes. the voltage dropped significantly, but the battery holder was not damaged. However, the plastic coating of the battery was discolored and partially melted. With the short circuit removed, the battery voltage recovered. A second test obtained similar results.

The staff reviewed the test results and finds that the lithium battery does not pose a significant ignition or explosion hazard. The SPEC 200 MICRO module continued functioning as designed even when the backup battery was short circuited. The SPEC 200 MICRO module does not need the battery for operation, but only for transport memory retention. With no backup battery, the module represents a channel trip when power is restored. With no power, a channel trip occurs due to the output signal going to a fail-safe condition. Thus, the staff concludes that the lithium batteries are acceptable as an auxiliary supporting feature.

5.10.2 Power Supplies

Each D. C. Cook RPS protection set will be equipped with new Foxboro power supplies. The transmitter instrument current loops will have 75 Vdc supplies. Each protection set will have 3 Foxboro Model PO300CQ power supplies. The

output of each power supply is diode-auctioneered to a common transmitter power supply bus. The failure of any one power supply does not affect the proper operation of the RPS protection set channels. Power to the power supplies is from the existing station 120 Vac power sources.

The signal processing rack mounted instrument modules have a similar power source arrangement. The power to these instrument modules is ±15 Vdc. Each RPS channel has 4 Foxboro Model 2ARPS05 linear, series-pass ±15 Vdc power supplies that are diode auctioneered to the nest power distribution buses. The failure of any one power supply does not affect the proper operation of the RPS channel. Power to the power supplies is from the existing station 120 Vac power sources.

The circuit breakers feeding the instrument (transmitter and rack mounted) power supplies will each be rated at 20 amperes, which is capable of powering all connected loads. However, due to the filter capacitors on the output of the power supplies, the inrush current may exceed 50 amperes if not controlled. This could cause nuisance tripping of the circuit breaker and transfer of the solid state transfer switch or current limiting of the inverter. To address this possibility, the design incorporates a 2.5 ohm resistor in series with the power leads. A time delay-upon-energizing relay operates contacts that bypass the resistor after 2 seconds. These components were purchased as Class IE. A control switch is be manually reset per operating procedures after energizing the channel to prevent a current inrush the next time the cabinet is energized. The staff reviewed the procedure entitled "Foxboro Protection Rack Inrush Current Control Function

Verification," 12IHP6030.IMP.015 Rev 0 to verify functionality of the inrush current limiting circuits, and finds it acceptable. Performance of this procedure assures that the inrush current control circuit performs its function properly. The licensee is developing procedures which will require I&C personnel to reset the current limiter and perform the functionality checks each time power is restored to the RPS cabinets. This is acceptable to the staff.

The existing station 120 Vac power sources have adequate capacity to provide power to the new Foxboro power supplies. Should the solid state transfer switch change from normal inverter power to the alternate regulated 120 Vac power, that transfer occurs in one-quarter cycle (less than five milliseconds). The cycle time of the microprocessor is 200 milliseconds. The filter capacitors in the dc power supplies will maintain power for this brief switching interval. No degradation of signals, configuration, or output occurs during this switching transient. For a longer power outage, such as when the alternate regulated 120 Vac is in service and there is a loss of offsite power, the lithium battery will maintain the configuration of the microprocessor. The clock cycle resumes when power is restored, and the outputs are driven to the state required by the configuration and the input signals. Loss of power to the solid state protection system, where the individual channel trip signals are combined into reactor trip signals is already a postulated event and not affected by this modification. Any required reactor trip would occur. The staff finds the power supply systems acceptable auxiliary supporting features.

5.11 CONFIGURATION MANAGEMENT

The Configurator application (software used to create the configuration data that is downloaded to the SPEC 200 MICRO control cards to specify the functions performed and the input and outputs to each function), developed by the vendor, was reviewed by the staff. This configuration software is not considered by the vendor to be safety-grade because it is utility specific software that does not run on the RPS operating equipment. It runs under MS-DOS, and its only product is data that can be otherwise verified. Despite these assertions, the staff audited the Configurator software development process in the same way the operating system firmware residing in firmware modules was audited because of the importance of configuration data to proper system operation. No differences in the requirements and design documentation, review, testing, or change control processes were noted. The only potential deficiency noted was a lack of regression testing. Specifically, when a new version of DOS is released, the configuration data should be evaluated to confirm that the new version of DOS does not affect the data. However, the licensee has agreed to freeze DOS at Version 5.0 thus precluding the need for regression testing.

The purpose of the Configurator is to create configuration files that can be downloaded to the appropriate control cards. The engineering information driving the configuration data was derived systematically by the vendor from current plant drawings. The vendor analyzed the original use of H-Line analog equipment and designed a set of control cards using Control Blocks to produce

the same functionality. The licensee then verified each of these new drawings to ensure that design equivalence was obtained.

Failure of the Configurator to properly store, verify, print, or archive data or the random occurrence of an undetected corruption of a data disk could lead to the downloading of incorrect configuration data to individual control units. To deal with this concern, the channel calibration parameters (configuration), including setpoints, are to be independently verified by periodic surveillance and post-maintenance testing following data downloading from the Configurator. A walkthrough of the procedures for using the Configurator was provided by the licensee. The staff reviewed the administrative controls to be placed on its use. Independent configurators are supplied for Unit 1 and Unit 2. Each unit has its own dedicated Configurator mounted in a lockable cabinet along with the applicable interface equipment and a printer. The Configurator cabinets are color coded and will be located in separate rooms near the I&C Maintenance Shop to ensure Unit 1 and Unit 2 integrity. A Personal Computer (PC) is installed in the cabinets for configuration use only and will have only the configuration software and current configuration database for its specific unit.

The cabinets are kept locked, except when in use. The keyboards are keyed separately and are also kept locked. The keys are in a locked key box with authorized access under administrative controls. When an Engineering Control Procedure (ECP) dictates that a change should be made to a control card, the individual responsible for maintaining the configuration will first update the

database on the PC Hard Drive using the Configurator, then print a hard copy of the changes on the printer installed in the cabinet. The exact same changes are then made on the other unit's Configurator PC Hardware Drive by accessing the backup database, implementing changes, and printing a hard copy of the changes. Also, two backup floppies are created after each configuration change is made. One floppy is sent to AEPSC Corporate Office and the other is stored at the plant under administrative control.

Two procedures govern the use of the Configurator for modifying the Foxboro control cards. The high level procedure, (D. C. Cook Plant Procedure MHI-6030 "Administrative Control Policy For Instrument Configurators," September 1993, Rev. 0), covers aspects of the Configurator control packages and applies at the Configurator Cabinet level. The low level procedure for the Foxboro Configurator (D. C. Cook Draft Plant Procedure 12IHP6030.CFG.002, "Configuration of Foxboro Spec 200 Micro Control Card") provides step by step instructions for accessing the right program, making changes, and verifying changes. The high level procedure has been approved but the low level procedure was in draft form during the audit. Both procedures contain the appropriate content and level of detail. Most importantly, both demonstrate the appropriate formality and commitment to control that is required. The staff finds that acceptable configuration management is provided.

5.12 HUMAN FACTORS CONSIDERATIONS

Indications of the status of the Foxboro digital RPS are provided to the operator. The SPEC 200 MICRO control cards have three indicator light emitting diodes (LEDs). The indicator to the right of the battery is red when the battery is low. The other two LEDs are immediately below the battery. The STBY/CONT LED is normally steady green and indicates normal operating conditions. A blinking green indicates a STANDBY condition. The FAIL indicator, is a steady red in FAIL condition, and a blinking red in ERROR STANDBY condition. It is normally off. The control cards are mounted in racks with doors. During normal operations, these indicator LEDs are not observable and only the green LED is on. The STBY/CONT and FAIL LEDs are of particular use during the downloading of the configuration data from the Configurator and in diagnosing unanticipated behavior during plant operation or surveillance testing. During regular surveillance activities, the plant personnel will open the cabinets to observe the battery indicator LED as well as the STBY/CONT and FAIL LEDs.

Regarding the potential for mixing up the control cards, the plant representatives indicated that only one card would be configured at a time, and that all configuration activities would be done at a dedicated workbench. Before new or modified configuration data can be downloaded to a control card, that card will be removed from the nest and rack and placed on a workbench where the stand-alone personal computer used as the Configurator and the display module are placed. The control cards are labeled and will be properly returned to their nest and rack before another card is pulled for configuration. Post-maintenance testing procedures, and administrative controls such as component, rack and nest identification labeling are also in

returned to their nest and rack before another card is pulled for configuration. Post-maintenance testing procedures, and administrative controls such as component, rack and nest identification labeling are also in place. The staff finds that the above practiced provides acceptable control of the configuration process.

5.13 SOFTWARE RELIABILITY

Software reliability, a measure of how well a software system provides the services expected by its users, is built-into a product through use of quality-assurance including verification and validation during the development process. Software metrics (methods of quantifying aspects of the process or products of software development) have been developed to aid in the measurement of software system characteristics (such as reliability, maintainability, and efficiency), processes, and documentation. Some key reliability metrics include mean time between failure, rate of fault occurrence, probability of failure on demand, and availability. Other metrics combine two or more of these basic metrics to model the reliability growth of a system. These metrics can be used to predict when a particular level of reliability is attainable.

The growth model used by the vendor to mathematically summarize the fault data collected regarding the SPEC 200 MICRO software was the Duane Growth - Error Rate of Change as summarized in a report dated February 18, 1987. The Duane

Growth model not only yields an estimate of the rate of change of errors, but estimates the time to nest error and the number of errors expected in the near future. It utilizes the cumulative detected software defect rate during testing and the accumulated central processing unit operational time. Two other reliability metrics were noted in the Qualification Compliance Report: the Software Error Discovery Rate and Software Error Pareto Profile. The results contained in this report indicate that the software is acceptably reliable.

Availability data, coupled with the fact that the last problem report was received in 1987, would also indicate that the SPEC 200 MICRO firmware constitutes a highly reliable software product. Vendor representatives presented data to the effect that over 30 nuclear facilities and 400 non-nuclear facilities have installed over 14,000 SPEC 200 MICRO modules. The staff finds that these data demonstrate that SPEC 200 MICRO modules are highly reliable, and are, therefore, acceptable.

5.14 GROUNDING

Electrical noise is an important consideration in analog and digital instrumentation as well as signal conversion systems. Proper grounding of instrumentation circuits and equipment minimizes the potential for electrical noise affecting system operation. The D. C. Cook Nuclear Plant grounding practices and equipment grounding methods were evaluated by Hurst Engineering in Report 2985-HEI-02, "Engineering Analysis of Grounding Issues," Revision 2,

December 15, 1992. Redundant ground return paths using #4/0 copper conductors are provided for each RPS instrument rack to the plant grounding mat. The grounding mat provides a common ground reference point throughout the plant. Redundant connections to ground are used, with precautions taken against forming ground loops. The RPS power supplies will be connected to operate with the output power ungrounded. The power supplies are each grounded to the rack ground bus bar. The staff finds that acceptable grounding provisions are provided.

5.15 ELECTROMAGNETIC AND RADIO FREQUENCY INTERFERENCE

Electromagnetic compatibility is the capability of an electronic system to operate at design levels of accuracy and efficiency in an environment where electromagnetic interference (EMI) or electrical noise exists. Similarly, radio frequency interference (RFI) is any spurious voltage or current that appears in an instrument circuit due to a radio frequency signal in the instrument environment. EMI and RFI can either obscure valid signals or generate non-valid signals. Instrument design should negate any internally generated interference and be resistant to external EMI and RFI.

The replacement RPS equipment will be mounted in existing RPS instrumentation racks in both control rooms at the D. C. Cook Plant. The control room environment was tested for the presence of electromagnetic fields and radio frequency signals (Report 2985-HEI-03, Preliminary EMI/RFI Evaluation, AEPSC Reactor Protection and Control System Replacement Project, Hurst Engineering,

Revision 0). EMI/RFI surveys were conducted in each control room. Sources of EMI/RFI were researched for areas within 50 miles of the plant. The evaluation of interference sources took credit for the shielding inherent in the control room construction (grounded steel rebar in the concrete walls, floor, and ceiling). The actual field measurements validated that assumption.

Supplemental EMI/RFI testing was performed by the license. The site-specific requirements for this EMI/RFI testing were established consistent with the general guidance on test methodology in Military Standard 461C, Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference, April 1, 1980, and Military Standard 462, Electromagnetic Interference Characteristics, Measurements of, July 31 1967, with later notices on applications. Surveys indicated slightly higher amplitude signals in the D. C. Cook, Unit 2 control room compared to the Unit 1 control room. Much of the EMI/RFI in the control rooms is generated and radiated from within the control room due to power supplies and lighting. Plant communication (radio) signals are attenuated by the control room structure. Use of radio transmitters inside the control room is prohibited and controlled by administrative procedures. The results of this supplemental testing were provided in Test Report 60214-94N, Rev. 1 "Main Control Room Mapping Data and Process Control System EMI." Satisfactory results for EMI were demonstrated.

Factory testing of the Foxboro SPEC 200 dual MICRO modules indicated that they emit RFI between 50 MHz and 200 MHz. At one meter from the surface of the control card, the field strength was less than or equal to 0.00316 volts/meter

throughout the frequency range. The control room existing EMI/RFI field strength peak is at 1.5849 volts/meter at frequencies below 50 MHz. Between 50 MHZ and 200 MHz, the measured field strength was between 0.0178 volts/meter and 0.0316 volts/meter. Because of these low noise values, the upgraded system is expected to have no undesired affects on the existing electronic equipment.

The factory test included both conducted and radiated susceptibility using test methods from MIL-STD-462. No anomalies were noted. The staff finds that the above described testing provided confidence in the capabilities of the installed configuration to be sufficiently immune to EMI/RFI, and, therefore, this issue has been acceptably addressed.

5.16 TESTING

Functional testing of the completed RPS modification is necessary to ensure that the system is designed and will function as intended. The licensee's overall testing program is summarized in Report 2985-BJB-01, Test Program Summary, December 10, 1992, Revision 0.

The licensee's overall testing program for the RPS upgrade includes factory acceptance tests, integration tests, installation tests, surveillance tests, and time response tests. All types of testing, except the surveillance tests, are discussed in the following paragraphs. Periodic surveillance testing is discussed in Section 5.6 of this report.

5.16.1 Factory Acceptance Tests

The staff examined the factory acceptance test procedure (TP-150 Revision 2, Foxboro Factory Acceptance Test Procedures for the Upgrade of the Reactor Protection Process Instrumentation), witnessed portions of the test and reviewed selected test data. System checkouts and tests are performed under Foxboro Engineering Operating Procedures (Foxboro Engineering Operating Procedure EOP-201 Revision D, "System Checkout and Test of Nuclear Orders with Commercial Grade Equipment," October 29, 1991, and Foxboro Engineering Operating Procedure EOP-202 Revision C, "System Checkout and Functional Test of Nuclear Orders with Qualified Equipment," October 29,1991). Foxboro had completed their factory acceptance test, prior to the staff review. However, personnel from D. C. Cook were repeating the test while the staff review was underway. The licensee's tests were conducted on the factory assembled instrument racks with the same physical layout as that for the designed for the installation in the D. C. Cook control rooms. The instrument modules and power supplies tested were identified as the equipment to be shipped to D. C. Cook Nuclear Plant.

The factory acceptance test verified the equipment serial numbers, including power supplies, and verified equipment operability before packaging for shipment. The test procedures were sufficiently detailed to provide test history, notation of and resolution of anomalies, and a record of the test steps taken, test personnel, test instruments used, and the test configuration. Field inputs were simulated using current generators. The

racks were wired to the design configuration for D. C. Cook. Testing verified the correctness of the test configuration, the time responses, and proper response to input stimuli. Verifications included workmanship and the physical condition of materials and wiring. The staff identified no unresolved anomalies as a result of the factory acceptance test.

5.16.2 Integration Tests

Integration tests are performed before installation, but after the equipment is delivered to the plant site. These tests ensure that the replacement RPS instrumentation will interface with the control system without adverse interaction. Portions of the control system are being upgraded concurrent with the upgrade of the RPS process instrumentation. The integrating tests call for wiring the upgraded portion of the RPS and the control systems together per design drawings. Anticipated interactions are then simulated, including power failures, ground faults, and instrument failures. The results are recorded and documented. Any anomalies are corrected before proceeding with installation.

Integration Test Procedure #2985-U21NT was reviewed and found acceptable by the staff. The test procedure was well written with clear instructions and detail. The licensee is required to provide a summary of any anomalies identified, and corrections made. This is acceptable to the staff.

5.16.3 Installation Tests

Installation tests provide, in an orderly fashion, verification of installation according to the engineering design package, instructions, and drawings. This testing includes continuity checks of the de-energized wiring, cable testing, and visual inspections and examinations versus the installation drawings. Power, grounding, field wiring, and instrument cable shielding are verified before initial energization of the installed equipment. Trip, alarm, isolated digital outputs, and isolated analog outputs are verified to respond to input stimuli. Some instrument calibrations occur concurrent with the installation testing. Functional installation testing assures plant interfaces have been implemented properly. The licensee stated that any anomalies discovered during the installation testing will be corrected before proceeding.

5.16.4 Time Response Testing

Time Response Testing is part of the current Technical Specification required surveillance testing. The time response is measured between a simulated perturbation in the process to the channel trip, and includes all upgraded Foxboro instrumentation. The licensee indicated that the current time response surveillance procedures do not require change to accommodate the upgrade. Thus, the existing surveillance procedures are the control documents governing the time response testing of the newly installed RPS equipment.

Report 2985-HEI-01, "Summary Report for Response Time Evaluation," Hurst Engineering, November 9, 1992, Revision 0, described the results of the response time testing. The worst case published module response times for the resistance-to-current module used for OT/dT and OP/dT was 2 seconds with 400 milliseconds added as worst case for the associated SPEC 200 MICRO module. The staff reviewed the time response data recorded for the factory acceptance test using those modules. For the TBX-4116 channel, the response times were 600 milliseconds to de-energize and 900 milliseconds to energize. For the TBX-411C channel, the response times were 612.5 milliseconds to de-energize and 750 milliseconds to energize. For both cases, the response times were within the acceptance criteria.

The licensee will continue to use the existing Channel Time Response Test Procedures to conduct periodic technical specification required response time testing of the new RPS. Response time for the new equipment will be verified to meet the technical specification allowable limits. The staff finds that the above described testing satisfactorily demonstrates the new RPS functional capabilty.

5.17 DEFENSE-IN-DEPTH

In order to establish an appropriate level of reliability, the protection system is to be designed using techniques such as 1) diversity in component design and function to the extent practical, 2) fail safe configuration, and

3) simplification of design such that normal operating, maintenance, and postulated accident conditions do not result in the loss of the protection function.

Staff concerns regarding digital systems which are addressed by appropriate defense-in-depth include digital system susceptibilities to existing plant environments, the commercial dedication of digital hardware and software, on-site expertise for problem recognition and troubleshooting, the potential for common mode failures introduced by software errors and unintended functions, and software/hardware interaction problems. The staff review verified that there is sufficient functional diversity to mitigate an accident or transient given a common mode failure of a component (hardware or software) due to these areas of concern. Therefore, the licensee was requested to provide an evaluation/analysis using the methodology described in NUREG-0493 "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection Systems" design basis to demonstrate that sufficient diversity exists to mitigate accidents and transients assuming a postulated common failure. If a safety function is disabled, then a diverse means, that is unlikely to be subject to the same common mode failure, is to be provided to perform either the same function or a different safety function. The diverse function may be performed by a non-safety system if a certain quality level is maintained. Manual actions from the control room are acceptable if the analysis shows that adequate time and information, not susceptible to RPS replacement hardware/software common mode failure, is available to the operator for diagnostics and plant shutdown.

By letter dated December 22, 1993, the licensee provided a response to the staff's concern regarding defense-in-depth and diversity in the digital system. The licensee performed a functional diversity assessment of each updated FSAR (UFSAR) event assuming a common mode failure of the software in the RPS upgrade modification. In this assessment, all the FSAR events for both Units 1 and 2 of the D. C. Cook Nuclear Plant were considered. A review was performed to divide the events into those that were potentially affected by the postulated common mode software failure and those that were not affected. The potentially affected events were then individually evaluated qualitatively in light of the FSAR analysis to ensure that adequate diverse mitigation capability is maintained.

In the licensee analysis, if the trip function is processed outside of the new digital reactor protection system, then the trip is assumed to be available. If the trip is processed by a function that is a part of the new digital equipment, then the trip/ESF function is assumed to be lost. However, for some functions, alternate indications and/or diverse alarms are available. The alternate indications alarms that are available to the operator to mitigate the transient were given for the worst case event. A discussion was also provided concerning the operator's response/impact on reactor safety.

As part of the defense-in-depth analysis performed by the licensee for the Foxboro digital RPS modification, the licensee evaluated diversity for the new system against the existing anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC) to ensure that the requirements of 10 CFR 50.62 for ATWS diversity continue to be met. The licensee's

analysis indicated that diversity between the RPS modification and the AMSAC system could not be demonstrated since both consisted of digital equipment manufactured by Foxboro. Consequently, the licensee replaced the AMSAC system with a vendor-specific designed system consisting of Taylor-Mod 30 digital equipment. This modification was done to meet the equipment diversity requirement of 10 CFR 50.62. The staff finds the licensees analysis and subsequent change to be consistent with 10 CFR 50.62 and, therefore, acceptable.

5.17.1 Common Mode Failure Analysis Results

The staff notes that various reactor trips are not affected by the installation of the new digital RPS equipment. Among these trips are neutron high flux and high rate trips, undervoltage and underfrequency trips, and reactor trip on turbine trip. For events that are affected by a postulated common mode failure in the new digital equipment, the operator will be alerted to the event by alarms and/or indication. The operator can then provide the appropriate manual action and enter into the emergency operating procedures.

The licensees analysis indicated that for a locked rotor event, the consequences could be more severe than currently analyzed due to the longer response time assumed for the required mitigating actuation given the failure of the digital RPS. The FSAR analysis for this event assumes an instantaneous seizure of a reactor coolant pump rotor. For this event the reactor trips on low RCS flow. A common mode failure of the new digital reactor protection

system would result in the loss of the low flow reactor trip signal. As a result, manual action by the operator is the only method available for mitigating this event.

The locked rotor event evaluation indicated that the loss of reactor coolant flow will increase the coolant temperature and increase pressurizer pressure due to the reduction in heat removal. The wide range reactor coolant system (RCS) pressure recorders will still be available to the operator. The pressurizer pressure will continue to rise, with the operator receiving a high pressurizer pressure deviation alarm at 2325 psia for Unit 2 and 2175 psia for Unit 1. The reactor trip on high pressure is also lost to the postulated RPS common mode failure. The licensee stated that diverse high pressure alarms will be available and will draw the operators attention to trip the reactor manually.

The licensee stated that the locked rotor event is similar to the loss of forced reactor coolant flow in one loop but is more severe in that total core flow is reduced more rapidly to a lower value (approximately 70% in 2 seconds). As the coolant heats up, a significant increase in pressure will occur (2590 psia peak for both units). This peak is shown to occur two seconds after the reactor trip at one second. The licensee stated that this pressure is less than 110% of design pressure (2750 psia). However, if a reactor trip is delayed for approximately 60 seconds, the licensee indicated that the above design pressure may be exceeded. The licensee's analysis took no credit for operation of pressurizer spray or for the opening of the PORV's. The licensee stated that as is the case with the loss of forced reactor

coolant flow, the analysis was performed with a positive moderator temperature coefficient of +5 pcm/degrees F. This is more limiting than the TS limit of 100% reactor thermal power. The licensee stated that this assumption is conservative and provides substantial margin throughout core life.

As Tavg is increased in the analysis, power also increases. The licensee stated that a more realistic beginning of cycle MTC would be -4 pcm/degrees F. Throughout core life, the MTC would decrease to -20 pcm/degrees F. The licensee stated that the feedback from the MTC would, therefore, tend to shut the reactor down rather than increase power in an actual event. The possibility that automatic rod control could occur which would withdraw rods has no impact on the event because the rods are essentially already fully withdrawn at full power. On the basis of the above, the licensee concluded that it is unlikely that pressurizer pressure would exceed 2750 psia and virtually impossible that it would exceed 3200 psig (the ASME Boiler and Pressure Vessel Code level C criterion).

In the licensee analysis, DNB is expected to occur. In the event of a delay in reactor trip of 60 seconds, the number of fuel rods in DNB would increase. The licensee's analysis again did not assume operation of the pressurizer sprays, nor opening of the PORV'S. The licensee stated that the available flow will prevent the core from degrading to a condition to where it cannot be adequately cooled after trip. The portion of the core that experiences DNB will heat up until the Doppler coefficient shuts it down. The licensee stated that it is not expected that fuel melt will occur, but cladding burst and oxidation are anticipated. The licensee stated that substantial core damage

is acceptable for this highly unlikely postulated ANS Condition IV event with massive multiple failures in accordance with the current plant design basis

As noted, in the evaluation of the locked rotor event operator response to manually trip the reactor is assumed within 60 seconds. However, several pressurizer pressure alarms can be expected within seconds of event initiation including RCS acoustic monitor flow alarms since the pressurizer safety valves can will lift. Therefore, the licensee indicated that operator response may occur in less than the 60 seconds assumed for this event.

The most likely cause of this event as indicated by the licensee is the failure of the reactor coolant pump (RCP) or motor. The operator is provided with additional alarms of RCP status including, bearing temperature high, lower bearing seal water temperature high, lower bearing cooling water flow low, upper oil pot level high or low, and lower oil pot level high or low. The licensee indicated that this additional information available to the operator will increase the likelihood that the operator will respond promptly to manually trip the reactor.

The calculated offsite dose for the locked rotor event for the D.C. Cook Vantage 5 fuel was 0.3 rem whole body and 3 rem thyroid. With a delay in operator response of 60 seconds core damage is increased significantly. The analysis indicated that 10 CFR Part 100 criteria will be met for this Condition IV event. In addition, the maximum hypothetical accident doses will substantially bound the locked rotor event doses.

In order to assess the adequacy of the licensee's defense-in-depth analysis, the staff weighed the safety implications of: (1) the diverse back-up actuation that will not meet the safety analysis timing requirements if a common mode failure of the digital RPS is assumed; (2) the credit that the licensee gives to diverse indication in the control room that would facilitate manual actions if a common mode failure is assumed; (3) the staff findings regarding the software V&V program; (4) the operating experiences with a similar RPS at the Haddam Neck Power Plant, and (5) a qualitative assessment for this application that it is unlikely an accident or transient coupled with a common mode software failure will result in an RPS failure into other than the preferred state. Based on this assessment, the staff finds that there is reasonable assurance that if a software common mode failure occurs, there is a diverse means to safely shutdown the reactor.

5.18 REACTOR TRIP AND ENGINEERED SAFETY FEATURES TRIP SETPOINTS

All Trip Setpoints and Allowable Values for Reactor Trip and Engineered Safety Features Trip Setpoints other than those associated with Neutron Flux were evaluated to ensure values remained consistent with existing Technical Specifications. Neutron Flux setpoints were not evaluated because the nuclear instrumentation was not included in the upgrade, therefore, no new nuclear instrumentation uncertainties were introduced.

The new setpoint calculations provided by the licensee were based on Instrument Society of America (ISA) draft recommended practice ISA-dRP67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-

Related Instrumentation," (Draft 10) and the Westinghouse Menu Driven Setpoint Calculation Program, WCAP-12741 (STEPIT).

The calculations based on ISA-dRP67.04, Part II, methodology were performed on all upgraded instrument loops, and included uncertainties obtained from manufacturer published data, qualification test reports, plant specific procedures, and plant specific practices. The calculations included the uncertainties identified in ISA dRP67.04, Part II. The staff finds the calculation methods to be consistent with the ISA recommended practice. The uncertainty values used in the calculations for the new Foxboro equipment were verified by the staff to be conservative by review of applicable reference documentation and test results, and , therefore, the setpoint methodology is acceptable.

6.0 CONCLUSION

The staff has reviewed the proposed Foxboro digital RPS upgrade for D.C. Cook, Units 1 and 2 with particular attention focused on those aspects specific to ensuring safety in digital-based instrumentation and control systems including software quality and reliability, configuration management, environmental qualification of hardware, and defense-in-depth provisions in the event of postulated failures.

Based on its review, the staff concludes that the proposed modification to the RPS is consistent with the current plant licensing design basis and will

provide appropriate RPS function as required. The staff, therefore, concludes that the Foxboro digital RPS modification at D.C. Cook is acceptable.

ENCLOSURE 2

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

FACILITY: DONALD C. COOK NUCLEAR PLANT

SUMMARY OF REVIEW/INSPECTION ACTIVITIES

The licensee proposed a modification to the RPS due to the increased maintenance required on the existing Foxboro H-Line system and the difficulty in obtaining qualified replacement parts. The upgrade involves the installation of Foxboro's Spec 200 and Spec 200 Micro line of digital microprocessor-based instrumentation. The objective of this review is to assess the adequacy of the proposed Foxboro Spec 200 and Spec 200 Micro modification with emphasis on replacement system features that differ from the previously approved design. Key issues are the specific design bases for the new system and the original system; potential hardware vulnerabilities and susceptibilities; software development, verification and validation, and configuration management; potential control/protection system interactions; system failure modes and effects; reliability and the developmental and operational history of this equipment.

NARRATIVE DISCUSSION OF LICENSEE PERFORMANCE - FUNCTIONAL AREA

The licensee performed a thorough and detailed evaluation of the proposed instrumentation changes and provided substantial and well written documentation to support the proposed change. The licensee demonstrated knowledge of the hardware and software functions and capabilities and the issues involved in ensuring safety of digital-based systems. The licensee's staff provided prompt response to staff requests and accommodated the staff's needs expeditiously during the onsite audit. Also, the licensee demonstrated improved project management and technical expertise in contrast to an earlier RPS Upgrade proposal.

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