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

TECHNICAL EVALUATION REPORT

DOCKET NO. 50-266

DATED: December 4, 1980

Licensee: Wisconsin Electric Power Co.
Type Reactor: W PWR
Plant: Point Beach Unit 1

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Engineering Support Section
Reactor Construction and Engineering
Support Branch, RIII

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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 April 18, 1980, September 12, 1980, October 30, 1980.

The site inspection was completed on April 24, 1980. [4/] Generic and site specific guidance was requested from IE/NRR headquarters.[5/]

Summary of Licensee Actions/Statements

The environmental qualification of a number of components could not be completely documented because of the unavailability of detailed equipment qualification records. However, licensee believes the components would perform their safety-related functions under postulated accident conditions. The components include solenoid valves, limit switches, level switches, diaphrams, o-rings, electrical conductor seal assemblies, and a limited amount of electrical cable. These components will be replaced with environmentally-qualified, equivalent equipment. Qualification of pressure and differential pressure transmitters will be accomplished by replacing the original transmitters with environmentally-qualified transmitters.

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.

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.

- Safeguards Actuation
- Main and Auxiliary Steam Isolation
- Main and Auxiliary Feedwater Isolation
- Containment Air Purification/Cleanup
- Containment Combustible Gas Control
- Accumulator
- Pressurizer Spray
- Power Operated Relief Valves
- Steam Dump
- Containment Radiation Monitoring
- Containment Radiation/Sampling
- Service Water
- Emergency Power
- Control Room Habitability
- Safety Equipment 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 of 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 Point Beach Unit 1 Nuclear Power Plant 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 15 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 thru 8.)

9/ Attachment 8.

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT 1 REF[10]	CONCUR ?
Ia 5	MOV	Limitorque/Peerless	SMB/Frame PL12G/Class B	RHR/CS (SI) Crossover	AUX	1day	—	—	100	—	$2x10^8$	40yr	1,16	No - R,S
Ia 6	MOV	Limitorque/Peerless	SMB/Frame PH56B/Class B	CS Dischg (SI)	AUX	1 day	—	—	100	—	$2x10^8$	40yr	1,16	No R,S
IIa * 19	OIL	American Oil	Industrial #35	SI Pp Mtrs.	AUX	1yr *	—	—	100 ‡	—	$1x10^7$ ‡	1yr *	16,20	Yes - M,R
Ia 20	Motor	W	TBDP/Frame 445TS/Thermalastic Epoxy Ins.	RHR Pump	AUX	1yr *	—	—	100	—	$2x10^8$	40yr *	10	yes - R
Ia 22	MOV	Limitorque/Reliance	SMB/Frame M46/Class B	Comp. Clg. to RHR HeX	AUX	1yr	—	—	100	—	$2x10^8$	40yr	1,16	No - R,S,Y
Ia 58	Motor	W	ABDP/Frame 504-US/Class B Thermalistic Epoxy	Cntmt Fan Cooler	CNT	1yr *	310	80	100	H_3BO_3 NaOH	$2x10^8$	40yr	2,10	No - R, W
IIa * 59	Grease	Chevron	BRB-2	Cntmt Fan Cooler	CNT	1yr *	310	80	100	H_3BO_3 NaOH	$2x10^8$	1yr •	2,10	No - R, W, M
Ia 79	Cable, Inst.	BIW	Bostrad 7, Twisted shielded Pair	—	CNT	1yr *	318	90	100	H_3BO_3 NaOH	$1.5x10^8$	40yr *	14,15	Yes - R
Ia 80	Cable, Inst.	BIW	Bostrad 7 Twisted, shielded Pair	—	AUX	1yr *	318/316 ‡	90	100	H_3BO_3 NaOH	$1.5x10^8$	40yr *	14,15	Yes - R
Ia 83	Cable Splice	Bechtel/Raychem	Bechtel Dwg. SK-E-165	—	CNT	1yr *	290	56/54 ‡	100	H_3BO_3 * NaOH	$2x10^8$	40yr	1,11	yes - R
Ib 1	Motor	W	TBDP/Frame 444TS/Class B Moist. Res.	Cntmt Spray	AUX	1yr *	—	—	100	—	$2x10^6$	40yr *	18	No - R, B
Ib 2	Motor	W	688.5-H/Class B Therm. Epoxy	SI Pp	AUX	1yr *	—	—	100	-	$2x10^8$	40yr *	10	Yes - R
Ib 3	MOV	Limitorque/Reliance *	SMB/Frame L56/Class B	Cntmt Sump Suct. Iso	AUX	1yr *	—	—	100	-	$2x10^8$	40yr	1,16	No - R,S,Z
Ib 7	MOV	Limitorque/Reliance	SMB/Frame P56/Class B	RHR Pp Vsl. Inj.	CNT	1day	$\bar{X}$300	60	100	H_3BO_3 * Na-OH	$2x10^8$	40yr	1,16	No - R,S
Ib 8	MOV	Limitorque/Peerless	SMB/Frame PH56F/Class B	Cold Leg SI	CNT	1day	$\bar{X}$300	60	100	H_3BO_3 * Na-OH	$2x10^8$	40yr	1,16	No - R,S,Z

[10] Attachment 9 * Engineering Analysis ‡ Margin

Pt. Beach 1

* Out of Position

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	TTI REF	CONCUR?[10]
Ib 21	Motor	W	ABDP/Frame 504 US/B Therm Epoxy	Comp. Clg	AUX	1Yr *	—	—	100	—	$2X10^6$	40Yr *	18	Yes-R, B
Ib 36	RTD (See V 38)	Foxboro	DB13U-226W	RHR Suct/Dischg.	AUX	1Yr *	—	—	100	—	$1X10^6$ *	40 Yr *	19	Yes-R, R, Y
Ib 39	Oil	American Oil Co.	Industrial 21	Comp. Clg Wat. Pp	AUX	1Yr *	—	—	100	—	$1X10^7$ *	1Yr *	16, 20	No-R (NOT M)
Ib 53	MOV	Limitorque/Peerless	SMB/Frame D2223/Class B	AF Turb Supply	AUX	1day	$\bar{x}$300 *	60	100	—	—	40Yr	1, 16	No-R, S, Z, H
Ib 60	Cable Splice	W	W Dwg. # 206C391	Cntmt. Fan Clrs.	CNT	1Yr *	310	80	100	H_3BO_3 NaOH *	$2X10^8$ *	40Yr *	1, 2	No-K, E
IIa * 71	Oil	American Oil	Rykon Ind. #15	—	AUX	1Yr *	—	—	100 *	—	$1X10^7$ *	1Yr *	16, 20	Yes-R, M
Ib 72	Cable-600V Power	Kerite	HTK-Ins. FR-Jkt	—	CNT	1Yr *	320	105	100	H_3BO_3 NaOH	$2X10^8$	40Yr	22	No-R, B
Ib 73	Cable-600V Power	Kerite	HTK-Ins. FR-Jkt	—	AUX	1Yr *	320	105	100	—	$2X10^8$	40Yr	22	No-R, B
Ib 74	Cable-Control	Kerite	FR-Ins FR-Jkt	—	CNT	1Yr *	300	54/50 ■	100	H_3BO_3 NaOH	$1.75X10^8$	40Yr	22	No-R, B
Ib 75	Cable-Control	Kerite	FR-Ins FR-Jkt	—	AUX	1Yr *	316/300 ▲	76.4/50 ▲	100	—	$1.75X10^8$	40Yr	22	No-R, B, H
Ib 76	Cable-5KV Power	Okonite	Okonex-Ins. Okoprene-Jkt.	—	AUX	1Yr *	—	—	100 *	—	$5X10^6$ *	40Yr *	24, 26	Yes-R
Ib 77	Cable-Inst	Okonite	Okotherm-Ins Okoseal-Jkt. TSP	—	CNT	1Yr *	346	113	100	H_3BO_3 NaOH	$2X10^8$ *	40 yr *	24, 25, 26	Yes-R
Ib 78	Cable-Inst	Okonite	Okotherm-Ins Okoseal-Jkt TSP	—	AUX	1Yr *	346	113	100	—	$2X10^8$ *	40Yr *	24, 25, 26	Yes-R
Ib 82	Cable-Control	Rome	—	—	AUX	1Yr *	316/300 ▲	52	100	—	$1X10^7$ *	40Yr *	23, 24	No-R, H, O
Ib 84	Penetration	Crouse Hinds/W	Welded Cannister	—	CNT	1Yr *	340	104	100	H_3BO_3 NaOH *	$2X10^8$ *	40Yr *	5, 6, 22, 25	No-R, B

[10] Attachment 9   * Engineering Analysis   ■ Margin   ▲ Not Qualified   Pt. Beach 1   * Out of position

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CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR ?
IIa 10	Transducer	Fisher Governor	546	Sp. Add. Suct	AUX	1Yr *	—	—	100*	—	1x10^6 *	10 yrs*	7, 8, 17	yes-A, R
IIa 24	Transducer	Fisher Governor	546	RHR Ht.Ex Outlet	AUX	1Yr *	—	—	100*	—	5x10^6 *	10 yrs*	7, 8, 17	yes-A, R
IIa 26	Transducer	Fisher Governor	546	RHR Ht.Ex Bypass	AUX	1Yr *	—	—	100*	—	5x10^6 *	10 yrs*	7, 8, 17	Yes-A, R, Y
IIa 68	Grease	American Oil	Amolith #2	—	AUX	1Yr *	—	—	100	—	1x10^7 *	1yr *	16, 20	Yes-R, M
IIa 69	Grease	American Oil	Amolith #1EP	MOV's	CNT/AUX	1Yr *	$\bar{x}$ 300*	60	100	H_3BO_3 * $NaOH$	2x10^7	1Yr *	1, 16, 20	Yes-H, M, R
IIa 70	Grease	Mobil Oil	#28	MOV's	CNT/AUX	1Yr *	$\bar{x}$ 300*	60	100	H_3BO_3 * $NaOH$	3x10^9	1Yr *	1, 16, 20, 21	Yes-H, M, R
IIb 64	Limit Switch	NAMCO	EA180-11302	Aux. Chrg Line Cnt Iso. (Ind)	CNT	30days	340	75	100	H_3BO_3 $NaOH$	2.04x10^8	7Yr	4	Yes-K, L, R No-T, U, X
IIb 65	Limit Switch	NAMCO	EA180-11302	Rad. Monit. Cnt Iso. (Ind)	CNT	30days	340	75	100	H_3BO_3 $NaOH$	2.04x10^8	7Yr	4	Yes-K, L, R
IIb 66	Limit Switch	NAMCO	EA180-11302	Cnt. Purg. Sup & Ex. (Ind)	CNT	30days	340	75	100	H_3BO_3 $NaOH$	2.04x10^8	7Yr	4	Yes-KLR
III	None	—	—	—	—	—	—	—	—	—	—	—	—	—
IVa	None	Refer to: V12, 13, 15, 17, 32, 33, 35, 40, 41, 42, 43, 45, 47, 49, 50, 51, 52, 55, 56	57, 67	—	—	—	—	—	—	—	—	—	—	—
IVb	None	—	—	—	—	—	—	—	—	—	—	—	—	—
V 11	Limit Switch	NAMCO	D2400X	Sp. Add. TK. Suct (Ind)	AUX	—	—	—	—	—	—	—	—	Yes-J
V 12	Press. Xmtr	Foxboro	611GM	SI PP Disch	AUX	1Yr *	—	—	100	—	1x10^7	—	1, 9	yes-G, P, Q
V 13	Flow Xmtr	Barton	333	SI PP Disch	AUX	—	—	—	—	—	—	—	—	Yes-G, P, Y

[19] Attachment 9   *Engineering Analysis

Pt. Beach 1

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POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR ? [10]
V 15	Flow Xmtr	Barton	295	Rx. Vsl. SI	AUX	—	—	—	—	—	—	—	—	Yes - G, P, Y,
V 17	Level Xmtr	Barton	332	Sp. Add. Tank	AUX	—	—	—	—	—	—	—	—	Yes - G, P, Q
V 18	Level Switch	Magnetrol	A153	Cntmt Sump B	CNT	—	—	—	—	—	—	—	—	Yes - C
V 28	Limit Switch	NAMCO	D2400X	RHR Ht. Ex. Outlet (Ind)	AUX	—	—	—	—	—	—	—	—	Yes - J,
V 30	Limit Switch	NAMCO	D2400X	RHR Ht. Ex. Bypass (Ind)	AUX	—	—	—	—	—	—	—	—	Yes - J, Y
V 32	Press Xmtr	Foxboro	611GM	RHR Pp Dischg	AUX	1 Yr	—	—	100	—	1×10^7	—	1	Yes - G, P, Q
V 33	Flow Xmtr	Foxboro	613DM	RHR Ht. Ex. Out	AUX	1 Yr *	—	—	100	—	1×10^7	—	1, 9	Yes - D, G, P, Y
V 35	Flow Xmtr	Foxboro	613DM	Comp. Clg. Ht. Ex. Out	AUX	1 yr *	320	90	100	—	—	—	1, 9	Yes - D, G, Y, P, Q
V 38	RTD	Foxboro	DB13U-226W	Comp Clg Ht. Ex Out	AUX	1 yr *	—	—	100 *	—	—	40 *	19	Yes - N, P, Q
V 40	Level Xmtr	Foxboro	613DM	Boric Acid Tank	AUX	1 yr *	320	90	100	—	—	—	1, 9	Yes - D, G, P, Q
V 41	Level Xmtr	Foxboro	613HM	Przr (RPS)	CNT	2 Hrs	290	60	100	H_3BO_3 NaOH	1×10^7	—	1	Yes - G, P, Q
V 42	Press Xmtr	Foxboro	611GM	Przr (RPS)	CNT	2 Hrs	290	60	100	H_3BO_3 NaOH	1×10^7	—	1	Yes - G, P, Q
V 43	Press Xmtr	Barton	332	Cntmt Press (RPS)	AUX	—	—	—	—	—	—	—	—	Yes - G, P, Y
V 45	Press Xmtr	Barton	332	Cntmt Press (RPS)	AUX	—	—	—	—	—	—	—	—	Yes - G, P, Y
V 47	Press Xmtr	Barton	332	Cntmt Press (RPS)	AUX	—	—	—	—	—	—	—	—	Yes - G, P, Y

[19] Attachment 9

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR?[10]
V 49	Press Xmtr	Foxboro	611GH	Rx. Clnt. Wd. Rng.	CNT	1Yr/3 Hrs ▲	290	60	100	H_3BO_3 NaOH	1×10^7	▲	1	Yes - D, F, G, P, Q
V 50	Level Xmtr	Foxboro	613HM	Stm. Gen. Nar. Rng (FW)	CNT	30 min	290	60	100	H_3BO_3* NaOH	1×10^7	▲	1	Yes - D, F, G, P, Q
V 51	Level Xmtr	Foxboro	613HM	Stm. Gen. Nar. Rng (FW)	CNT	30 min	290	60	100	H_3BO_3* NaOH	1×10^7	▲	1	Yes - D, G, P, Q
V 52	Level Xmtr	Foxboro	613HM	Stm. Gen. W. Rng (FW)	CNT	30 min	$\bar{X}$ 300	60	100	H_3BO_3* NaOH	1×10^7	▲	1	Yes - D, G, P, Q
V 55	Press. Xmtr	Foxboro	611GM	Main Steam	AUX	12 Hrs	—	—	100	—	—	▲	9	Yes - B, D, G, P, Q
V 56	Press. Xmtr	Foxboro	611GM	Main Steam	AUX	12 Hrs	320	90	100	—	—	▲	9	Yes - B, D, G, P, Q
V 57	Flow Xmtr	Foxboro	630	Main Steam	CNT	—	—	—	—	—	—	—	—	Yes - G, P, Q
V 61	Sol. Valve	ASCO	LBX831614	Aux Charg line ISO	CNT	—	—	—	—	—	—	—	—	No - T, U, X Yes - J
V 62	Sol. Valve	ASCO	HB8302C25	Rad. Monit. ISO	CNT	—	—	—	—	—	—	—	—	Yes - J, P, Q
V 63	Sol. Valve	ASCO	8320A7	Cntmt Purg Sup & Exh. ISO	CNT	—	—	—	—	—	—	—	—	Yes - J, P, Q
V 67	Level Xmtr	Foxboro	613DM	AFW COND. STORG.	TURB	1 day	320	90	100	—	—	—	—	Yes - G, P, Q
V 81	Cable, Control	Rome	—	—	CNT	1Yr/4 mos *▲	300	52	100	H_3BO_3* NaOH	1.45×10^{8} ▲* / 1×10^8	40*	17, 23, 24	Yes - O, P, Q

[10] Attachment 9   ▲ Not Qualified   * Eng. Analysis

POOR ORIGINAL

QUALIFICATION REFERENCES

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2 WCAP 7829, Fan Cooler Motor Unit Test, Westinghouse Electric Corp., Pittsburgh, PA., April, 1972.

3 WCAP 7343-L, Topical Report Irradiation Testing of Reactor Containment Fan Cooler Motor Insulation, Westinghouse Electric Corporation, Pittsburgh, PA., June, 1969.

4 Qualification of NAMCO Controls Limit Switch Model EA-180 to IEEE Standards 344 ('75), 323 ('74), and 382 ('72), Revision 1, ACME - Cleveland Development Co., Highland Heights, OH., March 3, 1978.

Estimation of Qualified Life of EA180 Series Nuclear Switch, Revision Dated Feb. 27, 1980, NAMCO Controls, Cleveland, OH.

Test Plan For the Qualification of Series EA180 and EA740 Switches For Use In Nuclear Power Plants In Compliance with IEEE Standards 323-74, 382-72, and 344-75, Revision 1, July 26, 1979, NAMCO Controls, Cleveland, OH.

Bechtel Letter From D. H. Clark to D. K. Porter, dated June, 1980, NAMCO Position Switches

5 Westinghouse Letter From R. L. Korner to W. F. Geisheker with the following Attachments, dated May 22, 1978, Qualification Data for the Point Beach Nuclear Power Plants Units #1 and #2;

1. PEN-RLK-3-16-01, Accident Environment Test Report
2. PEN-ACD-4-72-03, Accident Environment Test Report
3. ETL Report 5261 Reports of Seismic Tests on Electrical
4. ETL Report 5275 Penetrations for Westinghouse
5. Test Report on Incident Testing of Triax Penetration

WEPCO letter from R. L. Cantrell to T. J. Rodgers, dated March 1, 1974, Electrical Penetrations Point Beach Nuclear Plant.

WMPCo letter from R. L. Cantrell to Roger Newton, dated April 15, 1968, Electrical Penetrations.

WMPCo letter from R. L. Cantrell to A. A. Simmons, Project Manager - Westinghouse, dated September 9, 1968, Point Beach Nuclear Plant Electrical Penetrations.

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QUALIFICATION REFERENCES

6 Westinghouse Letter From R. L. Korner to W. F. Geisheker with Attachments, dated July 28, 1978, Point Beach Nuclear Plant Qualification of Containment Electrical Penetration Safeguards Splices.

1. PEN-TR-78-45, Boric Acid Effect on Medium Voltage Ceramic Seal-Bushing.
2. PEN-TR-78-11, Statement on Effect of Borated Water on Westinghouse Penetrations for the Angra Nuclear Plant.
3. Brunswick Nuclear Plant Drawing Nos. E-2457, E-2453, E-2452.
4. WEP, WIS Drawing Nos. 31402, 31396, 31400, 150-31393, 150-31394, 150-31396.

7 IE Bulletin 79-01B, Enclosure 4, Appendix C, Table C-1, Nuclear Regulatory Commission, Region III, Glen Ellyn, IL., Jan. 16, 1980.

8 Instruction Manual and Parts List, Fisher Controls Type 546 Electro-Pneumatic Transducer, Fisher Controls Co., Marshalltown, IA., Nov., 1968.

Fisher Controls Letter from Bill R. Flowers of W. D. Ehrke Co., Inc., to R. K. Hanneman, dated Sept. 29, 1980, Point Beach Nuclear Plant Environmental Qualification of Fisher Components.

9 WCAP-7354-L, Topical Report Supplier Post Accident Testing of Process Instrumentation, Westinghouse Electric Corp., Pittsburgh, PA., July, 1969.

10 Westinghouse Letter from C. A. Lins to R. K. Hanneman, dated June 2, 1980, Point Beach Nuclear Plant Bulletin 79-01B Motor Qualification.

WCAP8 5., Environmental Qualification of Class IE Motors For Nuclear Out-Of-Containment Use, Westinghouse Electric Corp., Pittsburgh, PA., June, 1976.

11 Westinghouse Letter WFP78-531, From R. L. Kelly to W. F. Geisheker with Attachments, dated June 28, 1978, Qualification of Containment Electrical Penetration Safeguards Splices.

Westinghouse Teletype PBW-B-3070 From N. E. Bush to J. K. Leslie of Bechtel, dated February 5, 1970, Splicing Information.

Bechtel Letter From H. E. Morris to W. F. Geisheker with Attachments, dated April 27, 1978, Point Beach Nuclear Plant Contatinment Electrical Penetration Splices:

1. Bechtel Drawing SK-E-165, Splicing Requirements for Penetration Lead Wires.
2. Bechtel Chronological List of Correspondence.
3. Bechtel Letter PBW-W 2789C From J. K. Leslie to W. B. Henderson of Westinghouse with Attachments, dated February 10, 1970, Penetration Splices.
4. Bechtel Letter From J. K. Leslie to W. B. Henderson of Westinghouse with Attachments, dated March 3, 1970, Penetration Splices.

QUALIFICATION REFERENCES

5. Westinghouse Teletype PBW-B-3179 From N. E. Bush to J. K. Leslie of Bechtel, dated March 5, 1970, Safeguard Cable Splices in the Containment.
6. Westinghouse Teletype PBW-B-3211, From N. E. Bush to J. K. Leslie of Bechtel, dated March 13, 1970, Containment Safeguards Splices.
7. Bechtel Letter PBB-W-2905 From J. K. Leslie to W. B. Henderson of Westinghouse, dated March 17, 1970, Splices for Safeguards Cables Inside Containment

12 Deleted

13 Deleted

14 Boston Insulated Wire & Cable Co. Letter dated April 23, 1980, from L. S. Lisker to R. K. Hanneman.

Report B901, BIW Bostrad^{7} and Bostrad^{7S} - Flame and Radiation Resistant Cables for Nuclear Power Plants, Boston Insulated Wire & Cable Company, Boston, MA., September, 1969.

15 Report IPS-348, Test Report - Steam Line Break/LOCA Exposure of Field Cables and Terminal Blocks For American Electric Power, Conax Corporation, Buffalo, N.Y., May, 1978.

16 Qualification Type Test Report, Limitorque Valve Actuators For Class IE Service Outside Primary Containment, Limitorque Corporation Test Laboratory, Lynchburg, Virginia, June 7, 1976.

17 Robert O. Bolt and James G. Carroll, California Research Corporation, Radiation Effects on Organic Materials, Richmond, California, Academic Press, New York, 1963.

18 Westinghouse Letter from C. A. Lins to R. K. Hanneman, dated June 2, 1980, Point Beach Nuclear Plant, Bulletin 79-01B, Motor Qualification.

Westinghouse Letter from C. A. Lins to R. K. Hanneman, dated August 29, 1980, Point Beach Nuclear Power Plant Equipment Qualification NRC Bulletin 79-01B Containment Spray Pump Motors Containment Fan Cooler Motors.

WEPCo Letter from R. K. Hanneman to C. A. Lins, dated September 8, 1980, Environmental Qualification of Containment Spray Pump and Component Cooling Motors at Point Beach Nuclear Plant.

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QUALIFICATION REFERENCES

19 Foxboro Letter from G. Tennesen to R. K. Hanneman, dated August 5, 1980, Resistance Temperature Detectors.

Installation Instructions and Parts List, "Dynatherm Resistance Bulbs with Aluminum Cap-Type Head, Model DB-1 Series", The Foxboro Company, January 1964.

20 Amoco Oil Company Letter from T. M. Warne, dated September 15, 1980, Radiation Resistance of Amoco Oil Lubricants; Project 4210, with attachment: The Effects of Radiation or Lubricants in Nuclear Generating Stations, by James S. Ferrie, Paul Leinonen, Dr. B. Neil, and E. Wharton (Ontario Hydro Research Division), for presentation at ASLE 35th Annual Meeting, Anaheim, California, May 1980.

21 Mobil Oil Letter from J. Kestly to R. K. Hannemen dated October 13, 1980, Radiation Information Wisconsin Electric Power, with attachments for radiation test data of Mobilgrease 28.

22 Kerite Letter from R. A. Olson to R. K. Hanneman, dated October 22, 1980, with attachment: Point Beach Nuclear Plant, Wisconsin Electric Power Company, LOCA Qualification of Kerite 600 Volt HTK Insulated, FR Jacketed Power Cable.

23 Rome Cable Corporation Letter from D. D. Sand to R. K. Hanneman, dated April 9, 1980.

Rome Cable Corporation letter from D. D. Sand to P. R. Belhumeur dated March 19, 1971.

24 Okonite Company Letter from J. S. Lasky to R. K. Hanneman, dated May 9, 1980, With Attachment.

Blodgett, R. B. & Fisher, R.G., "Insulations and Jackets for Control and Power Cables in Thermal Reactor Nuclear Generating Stations, IEEE Transactions on Power Apparatus and Systems, Volume PAS-88, No. 5, May 1969.

25 "Type Test Cable Qualification Program and Data for Nuclear Plant Designed Life Simulation Through Simultaneous Exposure", Franklin Institute Research Laboratories, Final Report F-C3694, Philadelphia, Pennsylvania, January 1974.

26 Lancaster, Ron, "Qualification of Safety-Related Equipment Used in Nuclear Power Generating Stations Including the Effects of Aging", Reliability Conference for the Electric Power Industry, 1980.

Carfagno, S. P. & Gibson, R. J., "A Review of Equipment Aging Theory and Technology", Electric Power Research Institute, Final Report NP-1558, Palo Alto, California, September 1980.

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

Jablonski

October 21, 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 RCF for

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 POINT BEACH UNIT 1 - DOCKET 50-266

Frank Jablonski has completed the inspection phase at Point Beach Unit 1 in response to IEB 79-01B. A walkdown was conducted on October 10, 1980 to inspect installed components associated with the systems listed on the attachment; all components located outside containment.

Observations:

Motors

Motors for residual heat removal and component cooling water, the Auxiliary Coolant System, did not have nameplate data identical to the submittal. Both were stamped insulation class "B", neither was stamped "Thermalastic Epoxy" as stated in the submittal. Special insulation qualities were defined in correspondence between licensee and manufacturer, however, the correspondence was not reviewed during the inspection; the submittal will be corrected. Respective Westinghouse model numbers TBDP and ABDP were accurate.

Motor Operated Valve

The operator for the valve supplying component cooling water to the residual heat removal heat exchangers was a Limitorque type SMB-00 with a Reliance motor, insulation class "B"; installation complementary with submittal.

Limit Switches, Transmitters

NAMCO Snaplock D2400X limit switches installed on the RHR heat exchanger outlet and by-pass valves were scheduled for replacement, because no qualification documentation was available.

Onsite Inspection
ATTACHMENT 2

Foxboro transmitter Model 611GM, used for measurement of RHR pump discharge pressure, was scheduled for replacement.

Foxboro transmitters Model 613DM, used for measurement of outlet flow from the RHR and CCW heat exchangers were scheduled for replacement because manufacturer specified modifications (MCA) had not been completed.

Resistance temperature devices nameplate data complemented the submittal data.

Transducers (Converters)

Fisher Controls converters type 546, used for controlling the RHR heat exchanger outlet and by-pass valves, were being reviewed for qualification. The converters were contained in a NEC Class 1, Group D enclosure.

Miscellaneous

All equipment locations were verified to be complete and accurate as stated in the submittal. Plant equipment identification numbers reported in the submittal were determined to be either as stated, different than stated, or non-existent. For example, transmitter and pump motor identification were as stated; motor and air operated valves were different; transducers, limit switches, and resistance temperature devices non-existent. Physical location of the latter components, that is those without plant identification, provided reasonable assurance of correctness. (i.e., equipment was that listed in the submittal.) Several typographical errors were also identified on those submittal sheets thus far reviewed.

Conclusion

Except as reported above, the equipment descriptions provided by the licensee on the system component evaluation worksheets for the systems identified were complete and accurate.

The licensee was made aware of the apparent discrepancies. A detailed review will be made by the licensee and the response amended.

D.W. Hayes

D. W. Hayes, Chief
Engineering Support Section 1

Attachment: As stated

cc: J. G. Keppler
G. Fiorelli
V. D. Thomas, IE:HQ
Resident Inspector

ATTACHMENT 2

AUXILIARY COOLING SYSTEM

All Components Outside Containment

Plant Identification	Generic Name
1-P10	Pump Motor (RHR)
1-P11	Pump Motor (CCW)
1-AC73	Valve Motor Operator (CCW
1-AC624	Electro-Pneumatic Transducer (RHR)
1-AC626	Electro-Pneumatic Transducer (RHR)
1-PT628	Pressure Transmitter (RHR
1-FT619	Flow Transmitter (RHR)
1-FT619	Flow Transmitter (CCW)
1-TE627	Temperature Element (RHR)
1-TE621	Temperature Element (CCW)

ATTACHMENT 2

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

dupe 8012310083

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, Enginee-ing 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 <u>specific</u> unit type including motor type and insulation class. Almost all licensees refer to the various test reports as qualification documentation for <u>all</u> 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

dupe 8012310089

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

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		

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^{o}	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

dupe 8008070229

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

THR 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

dupe 8008070232

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

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).

dupe 8008070238

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

dupe of 8008070241 8008070241 DUPLICATE

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

SEP 11 1980

DIR			
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RC&ES			
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Docket No. 50-266
-301

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

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

SUBJECT: LACK OF SEPARATION CRITERIA AT POINT BEACH (AITS F03059680)

Your request for a review of the Point Beach discussion of GDC-4 was located on pages 4.1-4 thru 4.1-5 of the FSAR and mislabeled as GDC-40. A copy is enclosed.

The response to your request for a definition of a high energy line is included in the enclosed memorandum to E. L. Jordan from D. G. Eisenhut, "Lack of Separation Criteria at Point Beach" September 3, 1980. This memorandum, also, outlines the NRR program on the review of pipe break criteria.

Action Item F03059680 is closed.

E L Jordan

Edward L. Jordan, Assistant Director
for Technical Programs
Division of Reactor Operations Inspection

Enclosures:
1. Point Beach FSAR pages 4.1-4&5
2. Memo D. G. Eisenhut to E. L. Jordan dated September 3, 1980

cc: J. Fair, NRR
RONS Regional Branch Chiefs

CONTACT: H. A. Wilber, IE
49-28180

Site Specific Issues
ATTACHMENT 3b

SEP 18 1980

dupe 8010080026

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated, and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture

of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

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SEP 3 1980

MEMORANDUM FOR: Edward L. Jordan, Assistant Director for Technical Programs, Division of Reactor Operation Inspection

FROM: Darrell G. Eisenhut, Director, Division of Licensing

SUBJECT: LACK OF SEPARATION CRITERIA AT POINT BEACH

REFERENCE: Memorandum, E.L. Jordan to D.G. Eisenhut, dated August 1, 1980.

In response to your request in the above referenced memorandum, we have reviewed the pipe break criteria that was presented and have investigated the current licensing activities in the area of pipe break criteria. You specifically requested that we review the interpretation of the criteria for a high energy system and provide a schedule for any proposed actions that may be necessary.

We have determined from our review of the pipe break criteria that the correct interpretation of a high energy line is a system where either the fluid temperature is greater than 200°F or the fluid pressure is greater than 275 psig. However, Appendix A of APCSB 3-1 defines a high energy fluid system as a fluid system that during normal plant conditions meets the temperature or pressure limits. Therefore, those portions of systems such as ECCS systems that do not exceed the temperature or pressure limits during normal operation would not be classified as high energy. This should resolve the Region IV concerns for the safety injection lines in the pump room. III New

As part of the NRC's Systematic Evaluation Program, the pipe break criteria for the SEP plants was reviewed. The results of this review revealed inconsistent application of the pipe break criteria for both inside and outside containment applications (see enclosed memo). As a result of this study a generic letter to the SEP licensees has been prepared to address the application of pipe break criteria inside the containment. A generic letter to all other licensees addressing pipe break criteria inside the containment is currently planned. Resolution of the pipe break criteria outside the containment is pending the results of the inside the containment reviews.

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

dupe 8010030050

Contact:
J. Fair, X27357

OFFICE ▶						
SURNAME ▶						
DATE ▶						

ATTACHMENT 3b

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

May 15, 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: LACK OF SEPARATION CRITERIA AT POINT BEACH (A/I F03059680)

Ref: April 24, 1980 memorandum to D. W. Hayes from F. J. Jablonski, same subject (copy attached)

Per our discussion, please advise us of the NRC position relative to the matter discussed in Mr. Jablonski's memorandum.

Our cursory review of the FSAR for Point Beach did not locate where overall plant requirements per GDC-4, "Environmental and Missile Design Basis", and GDC-5, "Sharing of Structures, Systems and Components" were discussed. However, in regard to the containment spray and safety injection pumps Figure 1.2-5 shows that these pumps for both Units 1 and 2 have a common location without separation.

In connection with the safety injection (SI) pumps, we would also like a clarification of what constitutes a high energy line. Our interpretation from Regulatory Guide 1.46 is that both a pressure above 275 psig and a temperature above 200°F must exist. Specifically, are the SI pump discharge lines which operate at about 1500 psig and less than 200°F considered high energy lines?

No further review or action on our part is planned pending receipt of your response.

DW Hayes

D. W. Hayes
Chief, Engineering Support Section 1

cc:
J.G. Keppler
G. Fiorelli
R.F. Heishman
R.F. Warnick
F.J. Jablonski

ATTACHMENT 3b

dupe 8008220250

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

May 15, 1980

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

FROM: F. J. Jablonski, Reactor Inspector

SUBJECT: POTENTIAL DISCREPANCY WITH CRITERIA USED IN THE SER OF HIGH ENERGY LINE FAILURE AT POINT BEACH RE: MEMO HAYES-JORDAN MAY 2, 1980

On May 14, 1980 I had a discussion with Mr. C.J. DeBevec, IE:HQ, regarding the above. Mr. DeBevec explained that Regulatory Guide 1.46 states only what a high energy piping system is not. Further, Mr. DeBevec's understanding of AEC meetings held several years ago about the same subject confirms no discrepancy exists with criteria used in the SER at the Point Beach Nuclear Power Plant. (Hi energy - where the temperature and pressure conditions of the fluid exceed $200^{\circ}F$ and 275 psig).

F. J. Jablonski

F. J. Jablonski
Reactor Inspector
Engineering Support Section 1

cc:
J.G. Keppler
G. Fiorelli
J. Smith
V.D. Thomas, IE:HQ
C.J. DeBevec, IE:HQ

ATTACHMENT 3b

dupe 8101080034

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

April 24, 1980

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

FROM: F. J. Jablonski

SUBJECT: LACK OF SEPARATION CRITERIA AT POINT BEACH

During my trip to Point Beach on April 21, 1980 relative to IEB 79-01B, I observed what appeared to be a total lack of separation criteria. NOTE: Only the SI system was observed. For example the SI and CS pumps for both Units 1 and 2 share a common room without any separation between redundant pumps of Unit 1 or 2, or between Unit 1 and 2. The same condition exists at a different elevation for the Spray Additive Tanks.

Another example is the use of a single penetration for the passage of redundant cables used for indication of containment sump level.

Separation is beyond the scope of IEB 79-01; therefore there is a need for a separate memo. I realize separation criteria may have been different 15 years ago; however, these observations should be documented.

F. J. Jablonski

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cc:
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ATTACHMENT 3b

dupe 8008220254

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