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

TECHNICAL EVALUATION REPORT

DOCKET NO. 50-331

DATED: November 19, 1980

Licensee: Iowa Electric Light & Power Company
Type Reactor: BWR
Plant: Duane Arnold

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Reactor Construction and Engineering
Support Branch, RIII

CONTENTS

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 guidelines 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 14, 1980, May 5, 1980, and October 31, 1980.

The site inspection was completed on April 14, 1980.[4/] Generic and site specific guidance was requested from IE/NRR headquarters.[5/]

Summary of Licensee Actions/Statements

Investigations by the licensee indicate that almost all components are either not subjected to harsh environmental service conditions or have qualification documentation. Components determined to have incomplete qualification documentation at the present time will be tested, shielded, relocated, or replaced as soon as possible. In most instances, these actions will be taken during the 1981 refueling outage, but in no event later than June 30, 1982. In each case, justification for continued operation of DAEC has been provided. In the equipment qualification charts, the specified radiation dose is the calculated bounding dose from gamma radiation as per the licensing basis of the Duane Arnold Plant. (Not the guidelines)

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.

- Engineered Safeguards Actuation
- Low Pressure Coolant Injection
- Containment Spray
- Radiation Sampling
- Combustible Gas Control
- Closed Cooling Water System
- Reactor Water Cleanup System
- Reactor Recirculation System

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 assurance all components installed at the Duane Arnold Energy Center are environmentally qualified and installation methods of environmentally qualified components would not contribute to the failure of such components during a potential accident.

Based on several components categorized as IVb, the information submitted by the licensee did not fully and completely respond to the Order for Modification of License DPR-49. However, the licensee did provide justification for continued operation.

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

A positive conclusion cannot be made until:

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

2. The 8 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 F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT I REF[19]	CONCUR?
36 Ia	MOV	Limitorque	SMB-000	Class H* Insulation	Stm Tunnel	3day 17HR 3HR	252° 340°	15 105	Saturated Stm	NA	$2X10^8$	40yrs	5,6	Yes-A, P*
37 Ia	MOV	Limitorque	SMB-2	Class H* Insulation	Dry-Well	3day 17HR 3HR	252° 340°	15 105	Sat. Stm	NA	$2X10^8$	40YRS	5,6	Yes-A, P*
38 Ia	MOV	Limitorque	SMB-2	Class H* Insulation	Dry-Well	3day 17HR 3HR	252° 340°	15 105	Sat. Stm	NA	$2X10^8$	40yRS	5,6	Yes-A, P*
39 Ia	MOV	Limitorque	SMB-2	Class H* Insulation	Dry-Well	3day 17HR 3HR	250° 340°	15 105	Sat. Stm	NA	$2X10^8$	40YRS	5,6	Yes-A, P*
53 Ia	Limit Sw.	NAMCO	EA-740	Gasket is good for 1YR	Stm Tunnel	3HRS 30days	340° 200°	—	100%	NA	$2.4X10^8$	See Q	7	No-A, Q
54 Ia	Limit Sw.	NAMCO	EA-740	Gasket is good for 1YR	Dry-Well	3HRS 30days	340° 200°	—	100%	NA	$2.4X10^8$	See Q	7	No-A, Q
80 Ia	Solenoid Vlv.	ASCO	NP831665E	—	Dry-well	3 HRS 30 days	346° 200°	110 10	Stm	NA	$1.5X10^8$	4.4yrs ?	10	No-A, R
81 Ia	Solenoid Vlv.	ASCO	NP8320A183E	—	Drywell Stm Tunnel	3 HRS 30days	346° 200°	110 10	STM	NA	$1.5X10^8$	4.4yrs	10	No-A, R
82 Ia	Solenoid Vlv.	ASCO	NP8323A36 V-AC	—	Drywell Stm Tunnel	3 HRS 30 days	346° 200°	110 10	STM	NA	$1.5X10^8$	4.4yrs	10	No-A, R
83 Ia	Solenoid Vlv.	ASCO	NP8323A36 V-AC	—	Drywell Stm Tunnel	3 HRS 30 days	346° 200°	110 10	Stm	NA	$1.5X10^8$	4.4yrs	10	No-A, R
84 Ia	Solenoid Vlv.	ASCO	206-832-2RG	—	Dry-well	3 HRS 30days	346° 200°	110 10	Stm	NA	$1.5X10^8$	4.4yrs	10	No-A, R
118 Ia	5KV Cable	KERITE	HT Kerite with NJ Jacket	—	Various	13 HRS 7days	325° 228°	82 4	Sat. Stm	NA	$1.2X10^8$	40yrs	15, 16	Yes-A
119 Ia	600V Power & Control Cables	OKONITE	—	—	Various	55min 100days	340° 212+°	104 20	100%	NA	$1X10^8$	40YRS	17	Yes-A
120 Ia	600V Shielded Cables	OKONITE	1/C #14, 4/C #14, 7/C #14	—	Various	55min 100days	340° 212+°	104 20	100%	NA	$1X10^8$	40YRS	17	Yes-A
121 Ia	Coaxial Cable	Raychem	—	—	Various	10 HRS 26days	357° 212°	70 10	100%	NA	$2X10^8$	40yrs	18, 21	Yes-A

[19] Attachment 9
* See Attachment 3a

Duane Arnold

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT 1 REF#	CONCUR ?[10]
Ia 122	Splicing Kits	RayCHEM	(V)CSF-N	—	Various	10 Hrs 26 Days	357° 212°	70 10	100%	NA	2×10^8	40yrs	19, 20	yes-A
Ib 7	Flow Indicating Sw.	Barton	289	■ RHR Rm 4.26×10^6 Rads	N.W. Corner Rm	—	—	—	—	NA	■ 3×10^6	—	3	Yes-E,G,I
Ib 8	Flow Meter	S.K. Instrument	20-9651-8550	■ STm Tunnel 1.54×10^7 Rads	STm Tunnel	—	—	—	—	NA	■ 1×10^6	—	—	Yes-E,G,
Ib 9	Flow Sw.	Barton	289	HPCI Bypass	HPCI Rm	—	148°	—	100%	NA	—	—	—	No-A,L,N,E
Ib 10	Flow XmTR	GE	555-111BCAA3PDG	CTMT 130. Signal LoFl Cond.	RX Bldg	—	—	—	—	NA	—	—	—	No-A,D,E,L
Ib 11	Flow XmTR	GE	555-111BCAA3PDG	—	HPCI Rm	—	148°	—	100%	NA	—	—	—	No-A,L,N,E
Ib 12	Heater	GE	SA356W020	—	STm Tunnel	—	—	—	--	NA	—	—	—	No-A,G,N,E
Ib 14	Level Sw.	Robertshaw	82938-6-1	—	HPCI Rm	—	—	—	—	NA	—	—	—	D No-A,L,N,E
Ib 15	Level Sw.	Robertshaw	83841-A2	Supp. Pool Hi Level	Torus Rm N	—	148°	—	100%	NA	—	—	—	D No-A,L,N,E
Ib 16	Level Sw.	Magnetrol	S15.0-751	Scram Dscd. Vol. Hi Level Rx Scram Sig	RX Bldg	—	—	—	—	NA	—	—	—	G No-A,I,D,N,E
Ib 18	MOV	Limitorque	SMB-0	Class B* Insulation	N.W. Corner Rm	—	—	—	—	NA	2.04×10^8	40yrs	4	No-A,I,E
Ib 20	MOV	Limitorque	SMB-0	Class B* Insulation	RX Bldg	—	—	—	—	NA	2.04×10^8	40yrs	4	No-A,I,E
Ib 23	MOV	Limitorque	SMB-00	Class B* Insulation	S.E & N.W Corner Rm	—	—	—	—	NA	2.05×10^8	40yrs	4	No-A,I,E
Ib 24	MOV	Limitorque	SMB-000	Class B* Insulation	S.E & N.W Corner Rm	—	—	—	—	NA	2.04×10^8	40yrs	4	No-A,I,E
Ib 24	MOV	Limitorque	SMB-2	Class B* Insulation	RX Bldg	—	—	—	—	NA	2.04×10^8	40yrs	4	No-A,I,E

[10] Attachment 9
■ Radiation is lower than Required.

Duane Arnold

* See Attachment 3a

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR ?[10]
32 Ib	MOV	Limitorque	SMB-4	Class B* Insulation	N.W. Corner Rm	—	—	—	—	NA	2.04×10^8	40 yrs	4	No-A,I,E
33 Ib	MOV	Limitorque	SMB-5	Class B* Insulation	RHR Vlv Rm	—	—	—	—	NA	2.04×10^8	40 yrs	4	No-A,I,E
54 Ib	Position XMTR	Limitorque	—	RHR Service Wtr Sys.	HPCI Rm	—	—	—	—	NA	—	—	—	No-A,C,D
55 Ib	Press. Differ. Indicating Sw	Barton	P288A	CTMT Atmos Cont.	NE Corner Rm	—	—	—	—	NA	3×10^6	—	3	No-A,I
56 Ib	Press. Differ Indicating Sw	Barton	289	RHR Rm ■ 4.26×10^6 Rads	NE* SE Corner Rm	—	—	—	—	NA	■ 3×10^6	—	3	Yes-A,G,E
59 Ib	Press. Sw.	Barksdale	B2T-M12SS	—	HPCI Rm	—	148°	—	100%	NA	—	—	—	No-A,D,N
60 Ib	Press. Sw.	Barksdale	D1T-A80	CTMT Atmos Cont	RX Bldg	—	—	—	—	NA	—	—	—	No-A,I,N,D
61 Ib	Press. Sw.	Barksdale	D2T-M18SS	—	RX Bldg	—	—	—	—	NA	—	—	8	Yes-A,I,D
62 Ib	Press. Sw.	Barton	289	CTMT Atmos Cont	RX Bldg	—	—	—	—	NA	3×10^6	—	3	Yes-A,I,
63 Ib	Press. Sw.	STATIC-O-Ring	12N-AA5	Auto Depress.	SE Corner Rm	—	—	—	—	NA	—	—	—	Yes-A,I,D
64 Ib	Press. Sw.	STATIC-O-Ring	5N-AA3	—	HPCI Rm	—	—	—	—	NA	—	—	—	Yes-A,I,D
65 Ib	Press. Sw.	STATIC-O-Ring	5N-AA3	—	SE Corner Rm	—	—	—	—	NA	—	—	—	Yes-A,I,D
66 Ib	Press. Sw.	STATIC-O-Ring	5N-AA3	Auto Depress.	SE* NW Corner Rm	—	—	—	—	NA	—	—	—	Yes-A,I,D
67 Ib	Press. Sw.	STATIC-O-Ring	5N-AA3	Auto Depress.	SE* NE Corner	—	—	—	—	NA	—	—	—	Yes-A,I,D
68 Ib	Press. Sw.	STATIC-O-Ring	6N-AA2	Isolation Signal	HPCI Rm	—	—	—	—	NA	—	—	—	No-A,I,D,M

[10] Attachment 9

■ Radiation is lower than required

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* See Attachment 3a

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT1 REF	CONCUR?
69 Ib	Press. Sw.	STATIC-O-Ring	6N-AA21JX95TT	—	HPCI Rm	—	148°		100%	NA	—	—	—	No-A,N,D,M
75 Ib	Solenoid Vlv.	ASCO	FT8321A5	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
76 Ib	Solenoid Vlv	ASCO	FT832A6	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
79 Ib	Solenoid Vlv	ASCO	HVA-90-405	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
85 Ib	Solenoid Vlv	ASCO	8302C25U	Receives $<4\times10^5$ ■	RWCU Rm	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
89 Ib	Solenoid Vlv.	ASCO	831665	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
89 Ib	Solenoid Vlv.	ASCO	831665	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
90 Ib	Solenoid Vlv.	ASCO	831665	Receives $<4\times10^5$ ■	RHR Rm	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
92 Ib	Solenoid Vlv.	ASCO	831665	Receives $<4\times10^5$ ■	RHR Rm	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
95 Ib	Solenoid Vlv.	ASCO	8317A29	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
96 Ib	Solenoid Vlv.	ASCO	8320A90HB	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
97 Ib	Solenoid Vlv.	ASCO	8320A90HB	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
99 Ib	Solenoid Vlv.	ASCO	8320-A6	Receives $<4\times10^5$ ■	NW Cente Rm	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
100 Ib	Solenoid Vlv.	ASCO	8323-22	Receives $<4\times10^5$ ■	RX Bldg	—	—	—	—	NA	■ 4×10^5	—	9	Yes-A,I,D
101 Ib	Solenoid Vlv	ATKOMATIC	31840		RX Bldg	—	—	—	—	NA	4×10^6	—	11	Yes-A,I,D

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[19] Attachment 9

■ Radiation Less than qualified

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CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR?[19]
103 Ib	Solenoid Vlv.	Target Rock	72V-001	—	HPCI Rm	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
104 Ib	Solenoid Vlv.	Target Rock	72V-001	—	NW Corner Rm	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
105 Ib	Solenoid Vlv.	Target Rock	72V-002	—	SE Corner Rm	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
106 Ib	Solenoid Vlv.	Target Rock	72V-003	CTMT Atmos Cont.	Rx Bldg	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
106a Ib	Solenoid Vlv.	Target Rock	72V-003	CTMT Atmos Cont	Torus Rm	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
107 Ib	Solenoid Vlv.	Target Rock	72V-004	CTMT Atmos Cont	Torus Rm	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
107b Ib	Solenoid Vlv.	Target Rock	72V-004	CTMT Atmos Cont	Rx Bldg	—	—	—	—	NA	$1X10^8$	40YRS	13	Yes-A,I
108 Ib	TEMP. Element	NECI	N145C3023	Leak Detection	Torus Rm	—	148°	—	—	NA	—	—	—	No-A,N,D
108a Ib	TEMP. Element	NECI	N145C3023	Leak Detection	RCIC Rm	—	148°	—	—	NA	—	—	—	No-A,N,D
109 Ib	TEMP. Element	NECI	N145C3023	Leak Detection	HPCI Rm	—	148°	—	—	NA	—	—	—	No-A,N,D
110 Ib	TEMP. Element	NECI	N145C3023 P051	Leak Detection	RHR Rm	—	148°	—	—	NA	—	—	—	No-A,N,D
111 Ib	TEMP. Element	NECI	N145C3224	Leak Detection	RWCU Rm	—	—	—	—	NA	—	* —	—	No-A,C,D
112 Ib	TEMP. Element	NECI	N145C3224	Leak Detection	RWCU Rm	—	—	—	—	NA	—	* —	—	No-A,C,D
113 Ib	TEMP. Element	Rosemount	104MA23 ABBB	Leak Detection	STM Tunnel	—	300°	1.8	100%	NA	$4X10^6$	* —	14	No-A,C,D
116 Ib	Turbine	Terry Turbine	GE Order No. 205AA872 Type CCS	HPCI Pump Drive	HPCI Rm	—	—	—	—	NA	—	—	—	No-A,I

[19] Attachment 9

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* Thermal aging @ 104°F.

POOR ORIGINAL

101

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP	PRESS	RH	SPRAY	RAD	AGING	ATT I REF	CONCUR?
Ib [117]	250V DC MCC	ITE Imperial Corp.	Series 9600	Pwr Supply	Rx Bldg	—	—	—	—	NA	—	—	—	yes-A,G,I,E
IIa														
IIb														
III														
IVa [58]	Press XMTR	GE	?	Modification NUREG	Rx Bldg	—	—	—	—	—	—	—	—	yes-S
IVa [13]	H-O Monitor Panel	ComsIP	K4	Modification NuReg	Rx Bldg	—	—	—	—	—	—	—	—	yes-S
IVa [17]	Level XMTR	GE	?	Modification NuReg	Torus Rm	—	—	—	—	—	—	—	—	yes-S
IVa [70]	Press. SW	?	?	Modification NuReg	Dry well	—	—	—	—	—	—	—	—	yes-S
IVa [73]	Radiation Elem.	GE	?	Modification NuReg	Dry well	—	—	—	—	—	—	—	—	yes-S
IVa [74]	Radiation XMTR	?	?	Modification NuReg	Rx Bldg	—	—	—	—	—	—	—	—	yes-S
IVa [77]	Solenoid Vlv.	ASCO	HB8302C25RU	—	HPCI Rm	—	—	—	—	NA	—	—	9	yes-~~E~~,G,I
IVa [78]	Solenoid Vlv	ASCO	HT8321A5	Prevent Release Isolation Vlv	Radwaste Tk Rm	—	—	—	—	NA	—	—	9	yes-~~E~~,G,I
IVa [86]	Solenoid Vlv	ASCO	8302C25RU	NSS/CTMT Isolation	Torus Rm	—	—	—	—	NA	—	—	9	yes-G,E,I
IVa [88]	Solenoid Vlv.	ASCO	8302C25RU	NSS/CTMT Isolation	Torus Rm	—	—	—	—	NA	—	—	9	yes-G,E,I

[19] Attachment 9

Duane Arnold

POOR ORIGINAL

CAT	DESCRIPTION	MANUCACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT 1 REF	CONCUR ?[19]
IVa 91	Solenoid Vlv	ASCO	831665	Prevent Release Isolation Vlv	HPCI Rm	—	—	—	—	NA	—	—	9	Yes-G,I,E
IVa 93	Solenoid Vlv.	ASCO	831665	CTMT ISO. NSS	Torus Rm	—	—	—	—	NA	—	—	9	Yes-G,I,E
IVa 94	Solenoid Vlv	ASCO	831665	CTMT Atmp. Cont	H&V Control Valve Rm	—	—	—	—	NA	—	—	9	Yes-$\overset{A}{G}$,I,E,D
IVa 98	Solenoid Vlv	ASCO	8320-A6	RHR SYS	Torus Rm	—	—	—	—	NA	—	—	9	Yes-$\overset{G}{A}$,I,E,D
IVa 114	Terminal Blocks	?	?	Cable Termination	dry well RWCU	—	—	—	—	NA	—	—	—	Yes-F,E
IVa 115	Terminal Blocks	?	?	Cable Termination	RWCU Rm	—	—	—	—	NA	—	—	—	Yes-F,E
IVb 1	Air Cooling Fan Motor	Westinghouse	TEFC Serial # 7302	Eng. Safeguard Rms H&V	HPCI Rm	—	—	—	—	NA	—	—	—	Yes-F,E,I
IVb 2	Air Cooling Fan Motor	Westinghouse	TEFC Serial # 7302	Eng. Safeguard Rms H&V	NW/SE Corner Rms	—	—	—	—	NA	—	—	—	Yes-F,E,I
IVb 40	MOV	Limitorque	SMB-0	DC-* Class B Insulation	HPCI Rm	—	—	—	—	NA	—	—	—	~~No~~-F,E,G,I
IVb 41	MOV	Limitorque	SMB-00	DC-* Class B Insulation	RWCU Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
IVb 42	MOV	Limitorque	SMB-00	DC-* Class B Insulation	RCIC Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
IVb 43	MOV	Limitorque	SMB-00	DC-* Class B Insulation	STM Tunnel	—	—	—	—	NA	—	—	—	No-F,E,G,I
IVb 44	MOV	Limitorque	SMB-00	DC* Class B Insulation	Torus Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
IVb 45	MOV	Limitorque	SMB-00	DC* Class B Insulation	HPCI Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
IVb 46	MOV	Limitorque	SMB-000	DC* Class B Insulation	HPCI Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I

[19] Attachment 9
* See Attachment 3a

Duane Arnold

POOR ORIGINAL

CAT	DESCRIPTION	MANUFACTURER	MOD/TYPE	NOTES	LOC	OP TIME	TEMP F	PRESS PSIG	RH	SPRAY	RAD	AGING	ATT I REF	CONCUR ?[19]
47 IVb	MOV	Limitorque	SMB-000	DC Class B* Insulation	RCIC ¿ Rms	—	—	—	—	NA	—	—	—	No-F,E,G,I
48 IVb	MOV	Limitorque	SMB-2	DC Class B* Insulation	RHR Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
49 IVb	MOV	Limitorque	SMB-2	DC Class B* Insulation	HPCI Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
50 IVb	MOV	Limitorque	SMB-3	DC Class B* Insulation	HPCI Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
51 IVb	MOV	Limitorque	SMB-3	DC Class B* Insulation	STM Tunnel	—	—	—	—	NA	—	—	—	No-F,E,G,I
52 IVb	Motor-Operated Blower	SIEMENS	2CH6 041-U	MSIV Leak Control	RX Bldg	—	—	—	—	NA	—	—	—	Yes-F,E,G,I
57 IVb	Press. XMTR	GE	50-552032HKZZ2	RHR SYS	SE RHR Corner Rm	—	—	—	—	NA	—	—	—	Yes-F,E,G,I
71 IVb	RHR Pump Drive	GE	5K6336XC213A	RHR Sys	SE RHR Corner Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
72 IVb	CS Pump	GE	5K6336XC229A	CS SYS	SE RHR Corner Rm	—	—	—	—	NA	—	—	—	No-F,E,G,I
102 IVb	Solenoid VLV	Automatic VLV Co	C5450-5	Auto Depress.	Dry-well	3 HRS 3 HRS 2 days	340° 320° 250°	—	—	NA	3×10^7	—	12	No-A,F,E
V														
IVa	Solenoid VLV	ASCO	8302C26	NSS/CTMT ISO VLV	Torus Rm	—	—	—	—	NA	—	—	9	Yes-E,G,I

[19] Attachment 9
*See Attachment 3a

Duane Arnold

Page 9 of 9

POOR ORIGINAL

LIST OF TEST REPORTS

1. GE Qualification Report F01 for Electrical Penetration Assembly dated April 30, 1971.

2. Letter by Mr. G. G. Sherwood of GE to Mr. D. G. Eisenhut of NRC dated December 2, 1977.

3. ITT Barton Report No. R3-288A-1 dated May, 1980 and letter of Mr. L. L. Blake, Jr. of ITT Barton to Mr. J. C. Hink of Bechtel dated June 10, 1980.

4. Limitorque Qualification Test Report No. B0003, dated May 28, 1976.

5. Limitorque Qualification Test Report No. 600376A dated May 13, 1976.

6. Franklin Institute Research Laboratories Final Report No. F-C3441 dated September, 1972.

7. NAMCO letter to Bechtel dated September 8, 1980 and vendor print 7884-APED-E57-1.

8. Barksdale Bulletin 730701-E dated 1979.

9. Letter No. BLIEG-80-378 of Mr. J. L. Hurley of Bechtel Associates Professional Corp. to Mr. Philip D. Ward of Iowa Electric Light and Power Co. dated August 4, 1980.

10. ASCO Test Report No. AQS 21678/TR Rev. A.

11. Atkomatic Valve Co. Inc. letter from Gary Spear to Jim Hurley dated May 28, 1980, and Report No. 21 by, "Radiation Effect Information Center" of Battelle Memorial Institute.

12. Plant Equipment Design Engineering Memo No. 126-62, test No. 4.

13. Letter by Mr. D. K. Vater of Target Rock Corp. to Mr. Ron Garris of Iowa Electric dated August 29, 1980, Target Rock Corp. Test Report No. 2375 dated September 26, 1979, Appendix C, Appendix I, and Target Rock Test Report No. 2302 dated May 9, 1979.

14. Letter from Mr. Dan Whalen of Rosemount Inc. to Mr. J. L. Hurley of Bechtel dated July 11, 1980.

15. Franklin Institute Research Laboratories Technical Report No. F-C2737 dated April 30, 1970.

16. Kerite Report No. EM-178A and B dated May 23, 1977.

17. Okonite Company Engineering Report No. 127, Revision 1, dated November 5, 1971.

18. Franklin Institute Research Laboratories Technical Report No. F-C4033-1 dated January, 1975.

POOR ORIGINAL

List of Test Reports
ATTACHMENT 1

19. Franklin Institute Research Laboratories Technical Report No. F-C4033-1 dated January, 1975.

20. Raychem Report on Aging Study WCSF Compound Report No. ERR 2001 dated August 10, 1978.

21. Raychem Draft, "Interim Report of Flamtrol Thermal Aging Study" Laboratory Report No. 5058 dated February, 1976.

POOR ORIGINAL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
[illegible] ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

April 14, 1980

POOR ORIGINAL

MEMORANDUM FOR: V. D. Thomas, Technical Programs, Division of Reactor Operations Inspection, IE:HQ

THRU: D. H. Hayes, Chief, Engineering Support Section 1

for

FROM: J. Hughes, Reactor Inspector, Engineering Support Section 1

SUBJECT: SCREENING REVIEW OF LICENSEE RESPONSES TO IEB-79-01B AND SUMMARY OF INSPECTION OF INSTALLED SYSTEM AT DUANE ARNOLD - FACILITY DOCKET NO. 50-331

We have completed our initial screening review of the Duane Arnold facility response to IEB-79-01B, and have completed the inspection phase of the system audit.

I conducted a walkdown on March 11-12, 1980 to inspect installed components associated with the Core Spray and RHR systems. Prior to the walkdown, the following components were selected for review:

Tag Number	Component	Drawing
ZS2142	Limit Switch	M-121 Rev 9
V2142	Manual Valve L.O.	"
ZS2143	Limit Switch	"
V2143	Manual Valve L.O.	"
CV2118	Testable Check Valve	"
SV2118	Solenoid	"
ZS2118A&B	Limit Switch	"
CV2138	Testable Check Valve	"
SV2138	Solenoid	"
ZS2138A&B	Limit Switch	"
MO1900	Limitorque Operator Valve	M-120 Rev 13
MO1908	Limitorque Operator Valve	M-119 Rev 15
ZS1906A&B	Limit Switch	"
ZS1907	Manual Valve L.O.	"
CV1906	Testable Check Valve	"
SV1906	Solenoid	"
CV2002	Testable Check Valve	M-120 Rev 13
SV2002	Solenoid	"
ZS2002A&B	Limit Switch	"
ZS2003	Manual Valve L.O.	"

Onsite Inspection Report
ATTACHMENT 2

Results of the inspection established that the installation of the selected components was in accordance with specifications and drawings. Nameplate data was consistent with the records and included serial number, tag number, motor type insulation class and ratings. Electrical cables will be reviewed later under generic components.

The inspector questioned the licensee relative to solenoid and limit switches for testable valves. The licensee stated that these valves were only operated (tested) during a refueling outage, and that during operation they were in the safe position; therefore, the solenoids and switches need not be qualified. Limit switches for normally locked open valves (manual type) also fall in this same classification. The inspector agreed with the licensee's position.

The inspector requested the licensee to determine if the field run junction boxes located inside the containment were pull boxes or connection boxes. The licensee stated that if the boxes contained splices/terminal blocks, they would establish that they were properly environmentally qualified.

J. Hughes
Reactor Inspector
Engineering Support Section 1

cc:
J.G. Keppler, RIII
G. Fiorelli, RIII
G.C. Wright, RIII
W.S. Little, RIII
RIII Files

ATTACHMENT 2

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

ATTACHMENT 3a

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

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

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 E11GM 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]

PAGE 1 OF ______
REPORT NO [illegible]
PREPARED BY [illegible]
CHECKED BY J.T. [illegible]
DATE [illegible]

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	373.3	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	1643	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	2354	2 hours	Unknown.
4-12-76	2347.5	10 days	Bearing problem; see Appendix C.
4-22-76	2354	5-1/2 hours	Unknown.
4-29-76	[illegible].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 shut-down because failure occurred late Fri[illegible] and parts not [illegible] until [illegible].

ATTACHMENT 3a

DATE	TIME ON HOUR METER	LENGTH OF DOWN TIME	REASON
5-9-76	3140	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.[illegible]
CHECKED BY J.T. [illegible]
DATE April [illegible], 1977

DATE	TIME ON HOUR METER	LENGTH OF DOWN TIME	REASON
9-8-76	5954.[illegible]	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 seperation 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

None

Site Specific Issues
ATTACHMENT 3b

1. Control Rod Drive System
2. Primary Containment Isolation and Nuclear Steam Supply Shutoff System
3. Main Steam Line Isolation Valve Leakage Control
4. High-Pressure Coolant Injection System
5. Automatic Depressurization System
6. Core Spray System
7. Residual Heat Removal System
8. Standby Gas Treatment System
9. Standby AC Power Supply
10. DC Power Supply
11. Residual Heat Removal Service Water System
12. Emergency Service Water System
13. Reactor Protection System
14. Reactor Core Isolation Cooling System (Alternative Use Only)
15. Engineered Safeguard Rooms Heating and Ventilating System
16. Control Building Heating and Ventilating System
17. Standby Diesel Generator Room Ventilation System
18. Emergency Service Water Pump Room Heating and Ventilating System
19. Intake Structure Heating and Ventilating System
20. River Intake System
21. Electrical and Control Panels
22. Pumphouse Drain Sump
23. Leak Detection Systems
24. Containment Atmosphere Control
25. Main Feedwater (Alternative Use Only)
26. Ancillary Components
27. Area Radiation Monitoring (Alternative Use Only)
28. Nuclear Boiler/Containment Systems
29. NUREG 0578 Modifications

Licensee's System List
ATTACHMENT 4

SYSTEMS LIST

GE BWR

1. Engineered Safeguards Actuation
2. Reactor Protection System
3. Containment Isolation
4. Main Steam Isolation
5. High Pressure Coolant Injection
6. Low Pressure Coolant Injection
7. Automatic Depressurization System
8. Core Spray
9. Containment Spray
10. Residual Heat Removal
11. Standby Gas Treatment
12. Emergency Power
13. Service Water
14. Radiation Monitoring
15. Radiation Sampling
16. Combustible Gas Control
17. CRD Hydraulic System
18. Closed Cooling Water System
19. Condensate and Feed Water System
20. Reactor Water Cleanup System
21. Standby Liquid Control
22. Reactor Recirculation System
23. Reactor Core Isolation Cooling

NRR Systems List
ATTACHMENT 5

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 Testing scheduled, but not complete.

b. Qualification Records search still in progress.

V. Equipment Not Qualified

LER

Categories
ATTACHMENT 6

LER's

None

UNRESOLVED GENERIC - SPECIFIC ISSUES

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

2. There are no unresloved specific issues.

Unresolved Generic - Specific
Issues
ATTACHMENT 8

A. Subject to review and approval by NRR/EQB of test reports of other documentation.

B. Meets or exceeds specified parameters#.

C. Engineer Analysis (Aging).

D. Engineer Analysis (Radiation).

E. Justification for Continued Operation.

F. Components which inadequate test data exist, will be tested, shielded, relocated, or replaced with suitable components, no later than June 30, 1982.

G. Components which will be replaced during the March, 1981 refueling outage or shielded.

H. Cannot evaluate (?).

I. This note in regard to consideration of qualified operating time for components which must be qualified for integrated radiation doses only following an accident. Qualification for pressure, temperature, and humidity is not applicable for these components during a high-energy line break because the component is not located within the same confined vicinity as a high-energy line.

Because the effects of integrated doses are cumulative and time or rate independent, operating time for these components is not applicable from an equipment qualification standpoint.

J. These Limitorque motor operators located in the rorus room have been qualified for operation at 250F for 24 hours and 200F for another 15 days. The temperature in the torus area will reach approximately 280F 7 seconds after the postulated HPCI steam line break, but falls immediately back down to approximately 200F. Based on the rationale that this high temperature lasts for only a very short duration and the fact that the prototype test sequence subjected the test specimen to an elevated temperature for a much longer duration (24 hours) than expected in actual service, the referenced test program is deemed to be adequate.

K. These Limitorque motor operators located in the steam tunnel have been qualified for operation at 250F for 24 hours and 200F for another 15 days. The temperature in the steam tunnel will reach approximately 300F immediately after the postulated main steam line break, but falls back down to approximately 200F in approximately 2 seconds. Based on the rationale that this high temperature lasts for a very short duration and the prototype test sequence subjected the test specimen to an elevated temperature for a much longer duration (24 hours) than expected in actual service, the referenced test program is deemed to be adequate.

#Beyond reviewer's expertise to determine specification adequacy.

Concurrence Code
ATTACHMENT 9

L. An analysis of the high-pressure coolant injection (HPCI) system operation under the postulated accidents defined by NRC IE Bulletin 79-01B has shown that HPCI system components are not subjected to a harsh environment for those accidents requiring the HPCI system to function.

M. These switches do contain Buna N diaphragm material. However, the expected radiation dose of less than 1 c 10^4 rad is well below the radiation susceptibility threshold level of 1 x 10^6 rad given by the NRC in Appendix C to Bulletin 79-01B.

N. GE has qualified these devices for operation at 148F during accident conditions. GE maintains that this temperature is the highest average compartment temperature expected to be seen where the device is located. This temperature is for the first hour and does not take into account temperature rise by direct steam impingement. Our calculations indicate that the temperature in these areas will exceed 148F during the first few seconds of the respective accident. However, GE amintains that its test philosophy is justified based on the following: all class 1E safety-related instruments are redundant and physically separated; four devices are generally available to provide the same function; and most sensing functions occur within the first few seconds of the onset of an accident and then are sealed in by control room relay logic.

O. These components were supplied by General Electric and have been qualified for operation at temperatures up to 212F. Calculations of expected temperature following a pipe break in these rooms indicate that the room temperature may rise to approximately 225F during the first few seconds. Also note that the temperature switches associated with these temperature elements have a setpoint of 130F. Therefore, actuation will take place before the temperature exceeds 212F in the room. Based on the reasoning that this temperature is a transient that lasts for a short period and that the elements would not reach the temperature of the surrounding ambient within this short period, these devices are sufficiently qualified.

P. See Attachment 3a, July 17, 1980 memorandum Jablonski to Hayes, "Motorized Valves".*

Q. It is intended to replace these gaskets on an interval consistent with their qualification (1 year).

R. Qualified for less than 40 years; no mention of program to verify replacement.

S. Procurement and installation is in accordance with requirements of NUREG-0578.