ENVIRONMENTAL QUALIFICATION

OF

SAFETY-RELATED ELECTRICAL EQUIPMENT

Big Rock Point Plant

October 1980

Consumers Power Company

Bechtel Associates Professional Corporation

Environmental Qualification of
Safety-Related Electrical Equipment

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SECTION I

INTRODUCTION AND BACKGROUND

On January 16, 1975, Consumers Power Company removed the Big Rock Point Plant from service after a design review of existing safeguards equipment revealed that they might be rendered inoperable to perform their function after the operation of a proposed new reactor depressurizing system. Many man-hours were expended during the subsequent six-month outage to qualify components in the two core spray systems, the two enclosure spray systems and the containment isolation system (inside containment). The results of the qualification effort as well as the modifications made to the Plant were provided to the NRC in Special Report No 21. The NRC indicated their agreement with the actions taken by Consumers Power Company to qualify the equipment in a letter from K R Goller to R B Sewell dated May 30, 1975.

On December 23, 1977, in a letter from V Stello to D A Bixel, the NRC requested that Consumers Power Company provide a submittal on the qualification of electrical equipment. The required response was provided on February 24, 1978, in a submittal from D A Bixel to D K Davis. An errata sheet to the February 24 1978 letter was submitted on March 23, 1978 in a letter from W S Skibitsky to D L Ziemann.

On September 6, 1978, Consumers Power Company was advised by NRC in a letter from D L Ziemann to D A Bixel that information transmitted in previous submittals had been put into a standard format. Consumers Power Company was requested to complete the formatted tables (as more information was required than was contained in previous submittals) and to confirm accuracy and completeness of the information. On November 30, 1978, Consumers Power Company provided the NRC the requested information in a letter from D A Bixel to D L Ziemann.

In a letter from D L Ziemann to D P Hoffman dated February 15, 1980, NRC provided Consumers Power Company a set of guidelines to use in performing a review of the qualification of electrical equipment. Subsequently, on August 29, 1980, the NRC transmitted an amendment to the Big Rock Point operating license requiring that Consumers Power Company submit by November 1, 1980 information necessary to support a safety evaluation report on the qualification of safety-related electrical equipment at the Big Rock Point Plant. This submittal contains the necessary information.

SECTION II

DEVELOPMENT OF THE EQUIPMENT LIST AND THE ENVIRONMENTAL PARAMETERS

The equipment list for the November 1, 1980 Big Rock Point Electrical Equipment Qualification effort was developed in three phases:

Phase I

1. Emergency Procedure EMP 3.3, Rev 12, "Loss of Reactor Coolant," was reviewed to obtain a list of equipment used by operators during the accident.

2. The list was shortened by deleting equipment that wasn't located in a hostile area. The hostile areas are defined in Section A.

3. Other equipment was deleted on a case-by-case basis as stated in Section B.

Phase II

1. The Instrument Data Book (Volume 4 of the Big Rock Point Plant Manual) was reviewed to obtain a listing of all equipment in the post-incident system, containment isolation system, the RDS system, the reactor protection system, the fire protection system, the emergency power system and the reactor vessel general system.

2. The list was then reduced based on whether or not the equipment was located in a hostile area, whether or not it was electrical and whether or not its function was deemed to be necessary to mitigate a LOCA or HELB.

Phase III

1. The lists obtained in the first two phases were then compared and the final list was generated. This list is found in Section C.

2. Criteria for determining required lifetime of the equipment subsequent to a LOCA or HELB event are documented in Section D.

3. Section E contains the environmental parameters for the five hostile areas.

A. Hostile Areas

Equipment considered for qualification was the equipment required to mitigate the effects of a LOCA or main steam line break (MSLB) inside containment or an MSLB outside containment and, in areas which suffer from the direct effects of these accidents, are defined as hostile areas. Areas of the Plant which house equipment required for the above accidents, but see only normal environment or a slight rise in room temperature due to operating equipment heat loss, are not hostile areas and are not considered here (as permitted per the NRC Regional Meetings on Electrical Equipment Qualification).

The five areas listed below are considered hostile for the purposes of electrical equipment qualification at Big Rock Point. Drawing 0740G40100 shows these areas.

1. The containment is subjected to pressure, temperature, humidity, gamma and beta radiation, submergence, containment spray and some thermal aging during the course of a main steam line break (MSLB) or a Loss of Coolant Accident (LOCA) inside.

2. The electrical penetration room is subjected to humidity during an MSLB outside of containment. During a LOCA inside containment, the electrical penetration room equipment receives a radiation dose due to shine from the containment wall. This room is unlike any of the other hostile areas outside of containment in that it is insulated and, therefore, suffers a temperature rise through the wall of containment.

3. The sphere ventilating room receives a gamma radiation dose due to shine from the containment wall during a LOCA. The construction of the building is such that heat can be readily dissipated; therefore, an ambient temperature is assumed. The structure is physically removed from the turbine building and will not be affected by an MSLB outside of containment.

4. The core spray room is a subterranean structure which houses the core spray pumps and core spray heat exchanger. The structure is physically removed from the containment wall and will, therefore, not be subjected to shine through the containment wall nor a temperature rise from heat transfer through the containment wall; however, when the containment sump water is recirculated through the heat exchanger subsequent to a LOCA, the recirculated fluid provides a gamma radiation source and a heat source. An MSLB outside containment has no effect on this room.

5. The pipe tunnel is located in the turbine building. It is an area where the fluid piping enters the containment building including the main steam and feedwater lines. Although some gamma radiation will be accrued by the equipment in the tunnel subsequent to a LOCA due to shine from the containment wall, the main steam line break provides a more severe environment to equipment in this area

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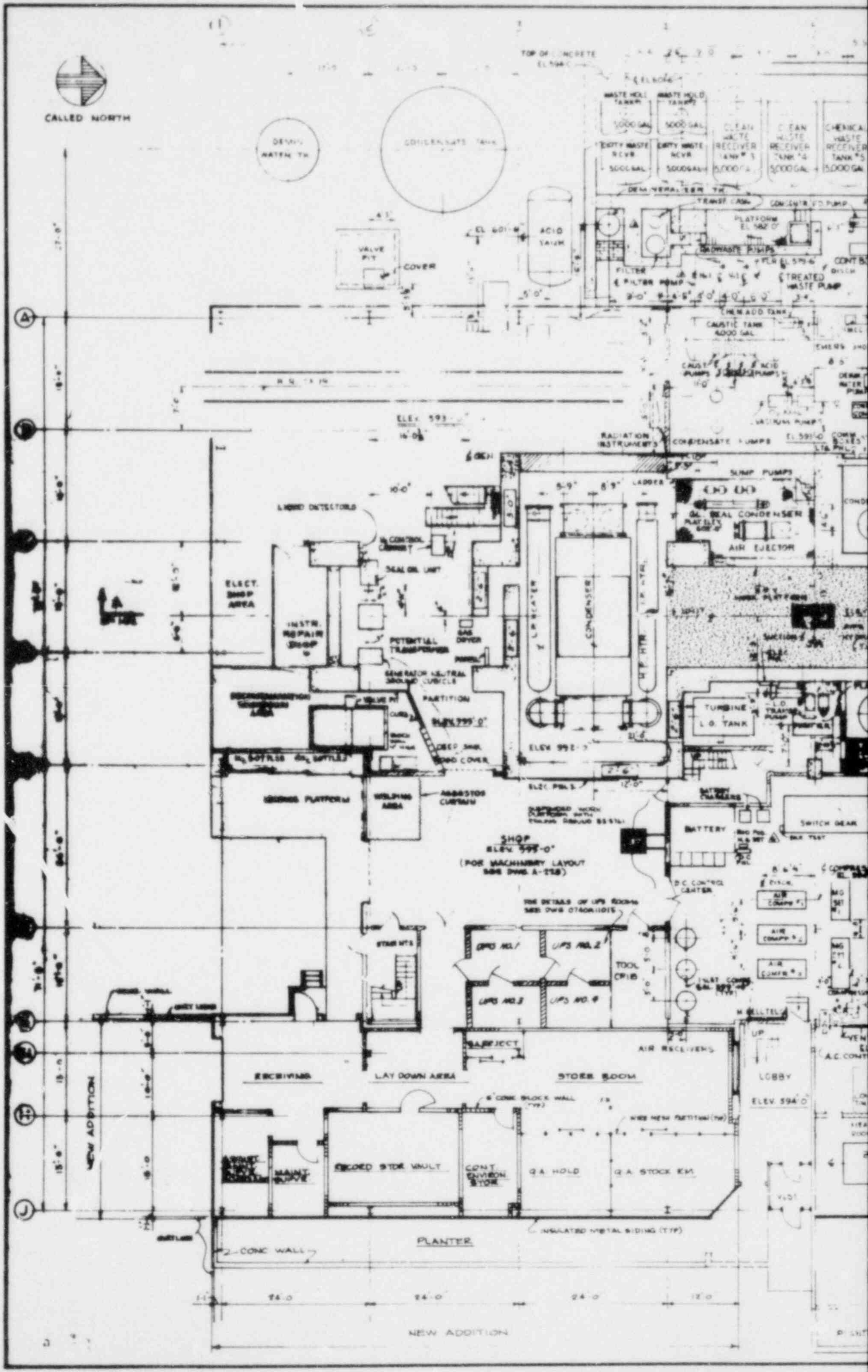

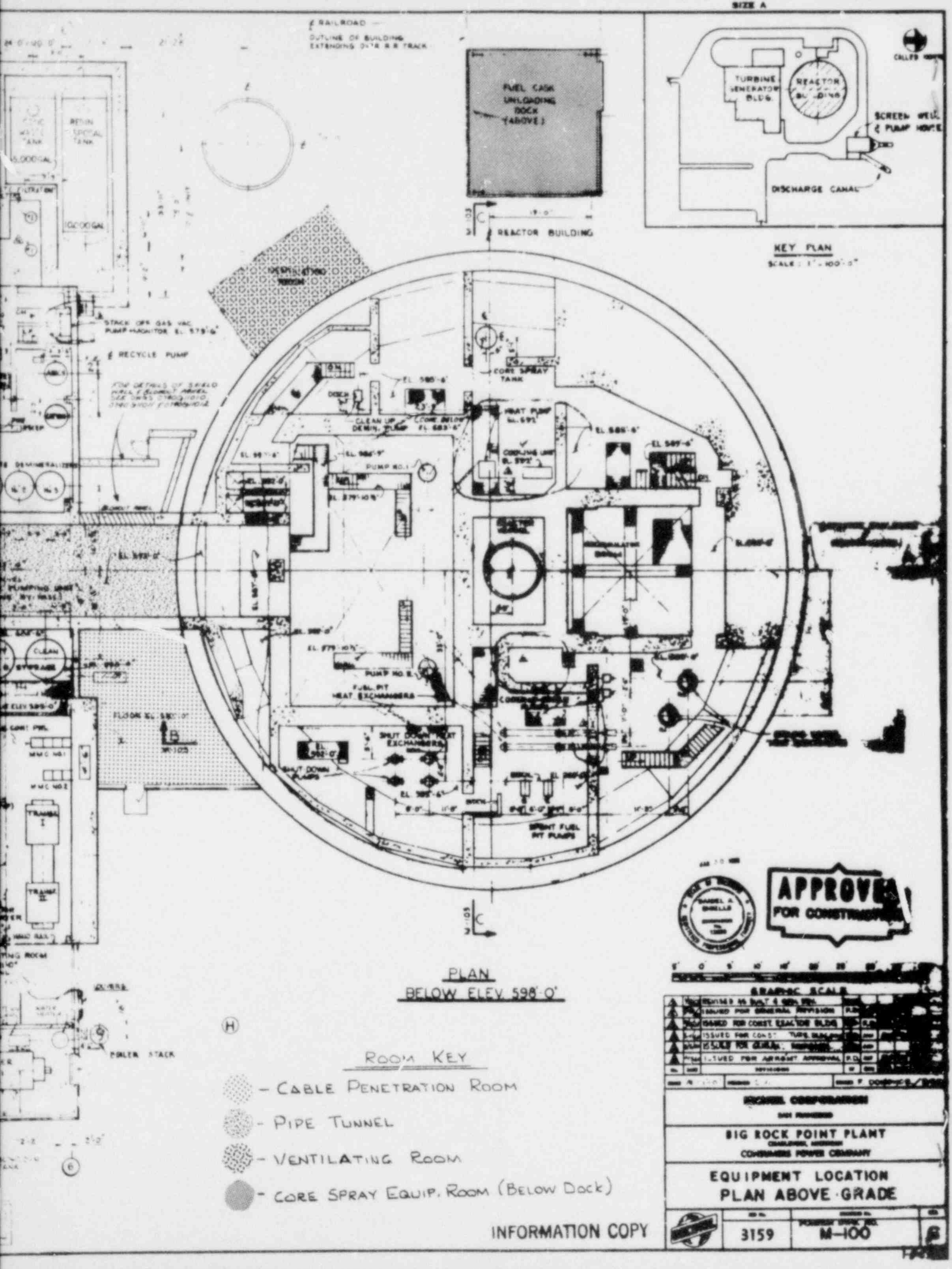

SIZE A
℄ RAILROAD
OUTLINE OF BUILDING EXTENDING OVER R.R. TRACK
FUEL CASK UNLOADING DOCK (ABOVE)
REACTOR BUILDING
TURBINE GENERATOR BLDG.
REACTOR BUILDING
SCREEN WELL & PUMP HOUSE
DISCHARGE CANAL
KEY PLAN
SCALE: 1" = 100'-0"
STACK OFF GAS VAC. PUMP-MONITOR EL. 579'-6"
℄ RECYCLE PUMP
CORE SPRAY TANK
CLEAN UP DEMIN. PUMP
CORE BELOW EL. 583'-6"
HEAT PUMP
COOLING UNIT
PUMP NO. 1
PUMP NO. 2
FUEL PIT HEAT EXCHANGERS
SHUT DOWN HEAT EXCHANGERS
SHUT DOWN PUMPS
SPENT FUEL PIT PUMPS
FLOOR EL. 590'-0"
MMC NO. 1
MMC NO. 2
TRANS. I
TRANS. II
BOILER STACK
PLAN
BELOW ELEV. 598'-0"
ROOM KEY
- CABLE PENETRATION ROOM
- PIPE TUNNEL
- VENTILATING ROOM
- CORE SPRAY EQUIP. ROOM (BELOW DOCK)
APPROVED FOR CONSTRUCTION
GRAPHIC SCALE
REVISED AS BUILT & GEN. REV.
ISSUED FOR GENERAL REVISION
ISSUED FOR CONST. REACTOR BLDG.
ISSUED FOR CONST. TURB. BLDG.
ISSUED FOR CLIENT
ISSUED FOR APPROVAL
REVISIONS
BECHTEL CORPORATION
SAN FRANCISCO
BIG ROCK POINT PLANT
CHARLEVOIX, MICHIGAN
CONSUMERS POWER COMPANY
EQUIPMENT LOCATION
PLAN ABOVE GRADE
3159
M-100
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INFORMATION COPY

B. Justification for Removing Equipment From Big Rock Point EEQ Listing

Equipment	Justification
MO-7069	This valve is electrically disabled in open position.
MO-7065	This valve is for the main steam line drain. It is a normally closed valve.
POS-6628 6629 6630 6649 6651 6656 6660 9101 9102 9103 9104 6627 6632 6618 6623 6624 6626 6653 6654 6655	NRC staff, during their site visit, stated that limit switches used for indication only need not be qualified at this time. All of these position switches are associated with containment isolation valves. The emergency procedure requires "subsequent" operator action to first check the vent valves closed and later to verify automatic isolation valves closed. If they don't close, the operator is to attempt to close them manually. This manual attempt probably would be done by the S-5 switch ("close penetrations and scram").
SV-4917	This solenoid valve controls isolation valve CV-4107 which is downstream of the main steam drain valve, MO-7065, which is normally closed. Since MO-7065 is normally closed, this valve is not required.
SV-4895 4858	These valves operate the control valves on the suction water to the control rod drive pumps. These valves are not needed since the pumps have integral backup isolation valves in addition to a check valve outside of containment. During the next refueling outage, a modification is scheduled to place a check valve in the suction line both inside and outside of containment.
SV-NC22C D	These solenoid valves control the scram dump tank isolation valves. They are excluded from qualification because their failure to function during a LOCA would allow leakage to the reactor building sumps, a negligible problem compared to the LOCA.

Equipment	Justification
I & C Power Panel 2Y	This panel is located in containment. It is fed from the 2B bus through I & C transformer 2B. A breaker located in the station power room, 22Y1, provides protection to the 2B bus and the other I & C preferred panels (1Y and 3Y) in the event 2Y fails.
RE-8258 RE-8259	These radiation monitoring instruments cause the containment vent valves to close due to radioactivity in the atmosphere. The intent originally was to have the valves close on high radiation in the event of a fuel handling accident rather than a LOCA. When thinking in terms of a LOCA, these devices become a backup to any of the scram input signals since any scram closes the vent valves. Pertinent scram inputs for this argument are low steam drum level or containment high pressure. In any larger size break LOCA, the radiation instruments will not have a chance to respond prior to the high containment pressure switches or the low steam drum level switches. In the very small size LOCA, the radiation levels may not be high enough to cause a vent valve closure via the radiation instruments. No credit has been taken for these instruments in mitigating the consequences of a small steam line break.
FE-5000 5001 5002 5003 5045 5046 CA-5000 5001 5002 5003 5045 5046	This equipment is used to provide acoustic monitoring on the steam drum relief valves. This equipment is presently scheduled to be qualified by Babcock and Wilcox. Also since this equipment is used for monitoring only, it can be qualified at a later time.

C. Equipment List

DEVICE IDENTIFICATION	RADIATION 100 % PWR MR/hr	RADIATION 100 % PWR R-40y	RADIATION SHUTDOWN MR/hr	RADIATION LOCA RAD	TEMPERATURE - °F 100% PWR	TEMPERATURE - °F SHUTDOWN	TEMPERATURE - °F LOCA	PRESSURE PSIA NORM	PRESSURE PSIA LOCA	RELATIVE HUMIDITY-% NORM	RELATIVE HUMIDITY-% LOCA	ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
FT-2161	2	7 E2	2	$7.3 \times 10^5 \gamma$ $1.32 \times 10^7 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	625.5	30 days	G 30045 G 30044
FT-2163	2	7 E2	2	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	625.5	30 days	G 40123
FT-2164	2	7 E2	2		50 - 90	50 - 90	235	14.7	41.7	20-80	100	625.5	30 days	G 40711 G 40712
FT-2162	10R/h γ,0.5 R/hrn	3.5E6	350		50 - 140	50 - 90	235	14.7	41.7	20-80	100	578	30 days	G 40713
LS-3562	3	1 E3	3	$4.9 \times 10^5 \gamma$ $4.2 \times 10^6 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	595	1 day	G 40123 G 30041
LS-3563	3	1 E3	3	$2.0 \times 10^6 \gamma$ $7.3 \times 10^5 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	587	1 day	G 30042
LS-3564	20	7 E3	5	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	579	1 day	
LS-3565	20	7 E3	5		50 - 90	50 - 90	235	14.7	41.7	20-80	100	574	1 day	
LS-RE09A	8	2.8E3	8	$2.0 \times 10^5 \gamma$ $1.3 \times 10^6 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	1 hour	G 30042
LS-RE09B	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	↓	G 40711
LS-RE09C	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5		G 40123
LS-RE09D	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5		
LS-RE09E	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5		
LS-RE09F	8		8		50 - 90	50 90	235	14.7	41.7	20-80	100	590.5		
LS-RE09G	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5		
LS-RE09H	8	2.8E3	8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5		
LT-3171 (Top)	3	1 E3	3	$7.3 \times 10^5 \gamma$ $1.32 \times 10^7 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	595	30 days	G 30042 G 40100
LT-3171 (Bot.)	20	7 E3	10	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	575	30 days	G 40123 G 40711
MO-7051	10R/hr γ .5R/hn	3.5E6	3.5×10^2	$7.3 \times 10^5 \gamma$ $1.32 \times 10^7 \beta$	50 - 130	50 - 90	235	14.7	41.7	20-80	100	599	30 days	G 40166 G 40711
MO-7061	10R/hγ .5R/hn	3.5E6	3.5×10^2	↓	50 - 130	50 - 90	235	14.7	41.7	20-80	100	599	30 days	G 40123

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DEVICE IDENTIFICATION	RADIATION 100 % PWR MR/hr	R-40y	SHUTDOWN MR/hr	LOCA RAD	TEMPERATURE - °F 100% PWR	SHUTDOWN	LOCA	PRESSURE PSIA NORM	LOCA	RELATIVE HUMIDITY-% NORM	LOCA	ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
MO-7064	2	7 E2	2	7.3×10^5 γ 1.32×10^7 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	625.5	30 days	G 30103 G 40165 G 40168
MO-7068	2	7 E2	2	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	622	30 days	G 40123
MO-7070	10	3.5E3	10	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	636.8	30 days	G 40123
MO-7071	10	3.5E3	10	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	636.8	30 days	G 40165
MO-7066	0.3	1 E2	0.3	4.0×10^4	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80	586	30 days	G 40123 G 40168
MO-7072	0.3	1 E2	0.3	4.0×10^4	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80	588	30 days	G 40711
PS-638	0.3	1 E2	0.3	4.0×10^4	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80	588	21 hours	G 40168
PS-7064A	4	1.4E3	0.1	2.0×10^4	40 - 100	40 - 100	182	14.7	41.7	20-80	20-100	597	1 hour	G 30029
PS-7064B	4	1.4E3	0.1	2.0×10^4	40 - 100	40 - 100	182	14.7	41.7	20-80	20-100	597	1 hour	G 40123
PS-IG11A	4	1.4E3	4	2.0×10^5 γ 1.3×10^6 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	1 hour	G 30045
PS-IG11B	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	1 hour	G 40123
PS-IG11C	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	1 hour	
PS-IG11D	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	1 hour	
PS-IG11E	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590	1 hour	
PS-IG11F	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590	1 hour	G 30042
PS-IG11G	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590	1 hour	G 40123
PS-IG11H	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590	1 hour	G 40711
PT-173	4	1.4E3	0.1	5.21×10^4 γ	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	599	30 days	G 30029
PT-187	4	1.4E3	0.1	↓	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	599	30 days	G 40123
PT-174	4	1.4E3	0.1	5.21×10^4 γ	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	599	30 days	
JB-160	8	2.8E3	8	2.0×10^5 γ 1.3×10^6 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	591	1 hour	G 30042
JB-161	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	591	↓	
JB-164	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	591	↓	
JB-166	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	591	↓	
JB-167	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	591	↓	

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DEVICE IDENTIFICATION	RADIATION				TEMPERATURE - °F			PRESSURE PSIA		RELATIVE HUMIDITY-%		ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
	100 % PWR		SHUTDOWN	LOCA										
	MR/hr	R-40y	MR/hr	RAD	100% PWR	SHUTDOWN	LOCA	NORM	LOCA	NORM	LOCA			
JB-180	4	1.4E3	4	$2.0 \times 10^5 \gamma$ $1.3 \times 10^6 \beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	1 hour	G 30042
JB-170	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-170A	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-170B	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-171	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-171A	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-171B	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	609	↓	
JB-IG11A	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	620	↓	
JB-IG11B	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	620	↓	
JB-IG11C	8	↓	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	620	↓	
JB-IG11D	8	2.8E3	8	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	620	↓	
Motor Starter (MO-7066)	0.3	1 E2	0.3	4.0×10^4	60 - 90	60 - 90		14.7	14.7	20-80	20-80	589	21 hours	G 40168
MO-7050	500 γ 10n	1.8E5	10	$2.0 \times 10^5 \gamma$ $1.3 \times 10^6 \beta$	50 - 150	50 - 90	235	14.7	41.7	20-80	100	612	10 min.	G 30043
MO-7067	500	1.4E5		$2.0 \times 10^4 \gamma$	50 - 130**	50 - 100	50-100	14.7	14.7	20-80	20-80	—	10 min.	
SV-4895	45**	1.6E4	45	$2.0 \times 10^4 \gamma$	50 - 100	50 - 100	50-100	14.7	14.7	20-80	20-80	Pipe Tunnel	30 days	G 40530
SV-4896	25**	8.8E3	25	↓	↓	↓	↓	14.7	14.7	20-80	20-80	Pipe Tunnel	↓	
SV-4922	30**	1.0E4	30	↓	↓	↓	↓	14.7	14.7	20-80	20-80	Pipe Tunnel	↓	G 30023
SV-4916	75**	2.6E4	75	↓	↓	↓	↓	14.7	14.7	20-80	20-80	Pipe Tunnel		
SV-4899	75**	2.6E4	75	↓	↓	↓	↓	14.7	14.7	20-80	20-80	Pipe Tunnel		
SV-4897	30**	1.0E4	30	↓	↓	↓	↓	14.7	14.7	20-80	20-80	Pipe Tunnel		
SV-9151	0.4	1.4E2	0.4	$1.9 \times 10^5 \gamma$	40 - 90*	40 - 90*	90	14.7	14.7	20-100	20-100	Vent Rm	30 days	G 30003
SV-9152	↓	↓	↓	↓	↓	↓	↓	14.7	14.7	20-100	20-100	Vent Rm	↓	
SV-9153	↓	↓	↓	↓	↓	↓	↓	14.7	14.7	20-100	20-100	Vent Rm	↓	
SV-9154	↓	↓	↓	↓	↓	↓	↓	14.7	14.7	20-100	20-100	Vent Rm	↓	

* Based on 1/78 thru 8/78 Rx Building temperature data - Supply air

** Measured 10/28/80, Reactor at power mwt

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DEVICE IDENTIFICATION	RADIATION 100 % PWR MR/hr	RADIATION 100 % PWR R-40y	RADIATION SHUTDOWN MR/hr	RADIATION LOCA RAD	TEMPERATURE - °F 100% PWR	TEMPERATURE - °F SHUTDOWN	TEMPERATURE - °F LOCA	PRESSURE PSIA NORM	PRESSURE PSIA LOCA	RELATIVE HUMIDITY-% NORM	RELATIVE HUMIDITY-% LOCA	ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
SV-9155	0.4	1.4E2	0.4	1.9 x 10^5 γ	40 - 90*	40 - 90*	90	14.7	14.7	20-100	20-100	Vent Rm	30 days	G 30003
SV-9156							90	14.7	14.7	20-100	20-100	Vent Rm		
SV-4891	3	3.5E2	3	7.3 x 10^5 γ	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590	30 days	G 30042
SV-4876	3	3.5E2	3	1.3 x 10^7 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	593		G 40533
SV-4869	3	3.5E2	3		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590		G 30044
SV-4879	5	1.8E3	5		50 - 90	50 - 90	235	14.7	41.7	20-80	100	613		G 40535
SV-4892	4	4.2E3	4		50 - 90	50 - 90	235	14.7	41.7	20-80	100	613		
LT-3180	8	2.8E3	8	7.3 x 10^5 γ 1.32 x 10^7 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	30 days	G 30042
LT-3181	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	30 days	G 30042
LT-3182	8		8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	30 days	G 30042
LT-3183	8	2.5E3	8		50 - 90	50 - 90	235	14.7	41.7	20-80	100	590.5	30 days	G 30042
LT-3184	4	1.4E3	4		50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	30 days	G 30045
LT-3185	4		4		50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	30 days	G 30045
LT-3186	4		4		50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	30 days	
LT-3187	4		4		50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	30 days	
SV-4980	14	4.9E3	14	7.3 x 10^5 γ 1.32 x 10^7 β	50 - 90**	50 - 90**	235	14.7	41.7	20-80	100	667	30 days	G 30045
SV-4981							235	14.7	41.7	20-80	100	667	30 days	G 41006
SV-4982							235	14.7	41.7	20-80	100	667	30 days	G 41007
SV-4983							235	14.7	41.7	20-80	100	667	30 days	
SV-4984				2.0 x 10^5 γ 1.3 x 10^6 β			235	14.7	41.7	20-80	100	665	1 hour	
SV-4985							235	14.7	41.7	20-80	100	665	1 hour	
SV-4986							235	14.7	41.7	20-80	100	665	1 hour	
SV-4987							235	14.7	41.7	20-80	100	665	1 hour	
Penetrations Inside														
H40	4	1.4E3	4	7.3 x 10^5 γ 1.32 x 10^7 β	50 - 90	50 - 90	235	14.7	41.7	20-80	100	612	30 days	G 30031
H65							235	14.7	41.7	20-80	100	606	30 days	
H80							235	14.7	41.7	20-80	100	602.5	30 days	
H81							235	14.7	41.7	20-80	100	602.5	30 days	

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* Based on 1/78 thru 8/78 Rx Building temperature data - Supply air

** Based on 1/78 thru 8/78 Rx Building temperature data - Em Cond area

DEVICE IDENTIFICATION	RADIATION 100 % PWR MR/hr	RADIATION 100 % PWR R-40y	RADIATION SHUTDOWN MR/hr	RADIATION LOCA RAD	TEMPERATURE - °F 100% PWR	TEMPERATURE - °F SHUTDOWN	TEMPERATURE - °F LOCA	PRESSURE PSIA NORM	PRESSURE PSIA LOCA	RELATIVE HUMIDITY-% NORM	RELATIVE HUMIDITY-% LOCA	ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
TB-240	14	4.9E3	14	$7.3 \times 10^{5}\gamma$ $1.32 \times 10^{7}\beta$	50 - 90*	50 - 90	235	14.7	41.7	20-80	100	~660	30 days	G 31006
TB-241	↓	↓	↓	↓	↓	50 - 90	235	14.7	41.7	20-80	100	~660	30 days	G 31024
TB-242	↓	↓	↓	↓	↓	50 - 90	235	14.7	41.7	20-80	100	~660	30 days	
TB-243				↓	↓	50 - 90	235	14.7	41.7	20-80	100	~660	30 days	
LS-RE06A	4	1.4E3	4	$2.0 \times 10^{5}\gamma$ $1.3 \times 10^{6}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	10 min.	G 30045
LS-RE06B	4		4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	10 min.	
LS-RE20A	4		4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	10 min.	
LS-RE20B	4		4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	621	10 min.	
PS-664	0.4	1.4E2	0.4	$2.0 \times 10^{4}\gamma$	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	599	1 hour	G 30029
PS-665	↓	↓	↓	↓	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	597	1 hour	
PS-666	↓	↓	↓	↓	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	599	1 hour	
PS-667	↓	↓	↓	↓	40 - 100	40 - 100	182	14.7	14.7	20-80	20-100	597	1 hour	
SV-NC22A	6	2.1E3	6	$2.0 \times 10^{5}\gamma$ $1.3 \times 10^{6}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	578**	10 min.	G 40533
SV-NC22B	6	↓	6		50 - 90	50 - 90	235	14.7	41.7	20-80	100	578**	10 min.	
SV-NC27 A5A thru F5B	1	3.5E2	1		50 - 90	50 - 90	235	14.7	41.7	20-80	100	580**	10 min.	
SV-NC22F	6	2.1E3	6	$7.3 \times 10^{5}\gamma$ $1.3 \times 10^{7}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	580***	30 days	G 40533
SV-NC22G	6	↓	6	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	580	30 days	
SV-NC22H	6	↓	6	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	580	30 days	
SV-NC22J	6	↓	6	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	580	30 days	
PT-IA07C	1	3.5E2	1	$7.3 \times 10^{5}\gamma$ $1.32 \times 10^{7}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	597	30 days	FC-407 Package
PDIS-7814	0.3	1 E2	0.3	$4.0 \times 10^{4}\gamma$ (a)	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80	587.75	21 hours	G 30029 G 40123 G 40714 G 40715
Core Spray #1	0.3	1 E2	0.3	$4.0 \times 10^{4}\gamma$	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80		30 days	
Pumps #2	0.3	1 E2	0.3	$4.0 \times 10^{4}\gamma$	60 - 90	60 - 90	152	14.7	14.7	20-80	20-80		30 days	

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* Based on 1/78 thru 8/78 Rx Building temperature data - Em Cond area
** Submerges (ie. 590')
*** Per R W Doan, SV-NC22F - J are located 1' above scram dump tank.
(a) This is a 30 day integrated dose.

DEVICE IDENTIFICATION	RADIATION				TEMPERATURE - °F			PRESSURE PSIA		RELATIVE HUMIDITY-%		ELEVATION FEET	TIME NEEDED	REFERENCE DRAWINGS
	100 % PWR		SHUTDOWN	LOCA										
	MR/hr	R-40y	MR/hr	RAD	100% PWR	SHUTDOWN	LOCA	NORM	LOCA	NORM	LOCA			
Panel C30 Terminal Blocks	4	1.4E3	4	$2.0 \times 10^{5}\gamma$ $1.3 \times 10^{6}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	616	1 hour	F 30910
Terminal Blocks behind instruments on steam drum wall	4	1.4E3	4	↓	50 - 90	50 - 90	235	14.7	41.7	20-80	100	620	10 min.*	F 30910
Penetrations	0.4	1.4E2	0.4	$7.3 \times 10^{5}\gamma$ $1.32 \times 10^{7}\beta$	50 - 90	50 - 90	235	14.7	41.7	20-80	100	≥598	30 days	
Core Damage Monitor Probe	4	1.4E3	4	$5.21 \times 10^{4}\gamma$	40 - 100	40 - 100	182	14.7	14.7	20-100	100	——	30 days	
Miscellaneous Electrical:	All conditions are the same as those associated with the instrument it connects with.													
Cable														
Splices														
Terminal Boxes														
Terminal Blocks														

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* For LS-RE06 and LS-RE20

D. Reasons for "Time Needed" for Big Rock Point EEQ

1.	FT-2161 2162 2163 2164	30 Days	These transmitters provide signal for flow in the two core spray and two enclosure sprays systems. Normal temperature and pressure are reached after three days, however, according to the LOCA temperature-pressure envelopes generated.
2.	LS-3562 3563 3564 3565	21 Hours	These switches provide indication for containment flood level. For a large break LOCA, it has been calculated that it will take no more than 20.8 hours to fill containment to 590' elevation. After it has filled, the operator goes to the recirculation mode and no longer needs this indication.
3.	LS-RE09A B C D E F G H	1 Hour	This time is based on the time it takes for low reactor water level to be reached when considering the smallest break analyzed for RDS actuation (ie, 0.008 ft^2).* Smaller breaks may result in later actuation of core spray but the containment environment up until the time of RDS/core spray will be much less severe than the qualification envelope. Further, radiation levels will be negligible until the time of RDS/core spray because the core will still be covered. The equipment is necessary for core spray valve open permissive and for containment isolation. Level indication for a period of 30 days can be obtained from LT-3180 through LT-3187.
4.	LT-3171	30 Days	See Item 2 above. This transmitter provides containment sump level indication. The time needed is 30 days due to TMI lessons learned.
5.	MO-7051 7061 7070 7071 7064 7068	30 Days	These valves are the core spray, backup core spray, enclosure and backup enclosure spray valves. At the latest, the first four valves will receive the open signal within one hour based on the discussion of Item 3. The last two valves are expected to be used intermittently for a period of one day, thereafter remaining in a closed position.

*Reference: SAFEC/BWR 1
5S228 Case 4
Date: 06/06/75
0.008 ft^2 Steam Break
RDS Time = 2366.5 Seconds
Core Spray Time = 2523 Seconds

5.5	MO-7067	10 Min	This is the turbine bypass isolation valve which closes on an isolation scram. It must then remain closed.
6.	MO-7066 7072	30 Days	These valves are positioned just before recirculation mode starts. The longest time calculated to fill containment to the highest flood level is 20.8 hours. See also Item 2. These valves see no adverse environment until they are positioned. They must only maintain their position for 30 days.
7.	PS-638	21 Hours	This pressure sensor provides an alarm indicating that the core spray pump either did not start or has no output (ie, it measures discharge pressure). Change from injection mode to recirc mode has been calculated to be 20.8 hours. This sensor sees no adverse environment until the recirculation mode begins and then is no longer needed.
8.	PS-7064A B	1 Hour	These pressure switches actuate the enclosure spray 15-minute timer. The electrical scheme indicates that the switches are not required thereafter. The longest time for containment pressure is after RDS blowdown. Therefore, time required is one hour. See Item 3.
9.	PS-IG11A B C D E F G H	1 Hour	These pressure switches provide a low reactor pressure permissive to the core spray valves allowing them to open. These switches are only needed as long as the LS-RE09A - H level switches. See Item 3. Time required is one hour.
10.	PT-173 187	30 Days	The pressure transmitters are used to indicate containment pressure from + 0.25 psig to - 1.0 psig. In addition, they are used to actuate valves for vacuum relief of containment.
11.	PT-174	30 Days	This pressure transmitter is used to indicate containment high pressure. It provides the operator information necessary to spray containment. The 30-day time is due to TMI lessons learned.
12.	MO-7050	10 Min	This is the main steam isolation valve. This valve closes automatically on low reactor water level or high containment pressure or loss of ac.

			The operator is instructed in emergency procedure EMP 3.3 to close the valve if drum level is falling uncontrollably. Ten minutes is considered to be a reasonable length of time for operator action to manually close the valve.
13.	SV-4895 4896 4922 4891 4876 48[illegible]9 4879 9155 9156	30 Days	These solenoid valves close and stay closed on receipt of a containment isolation signal from LS-RE09s or PS-664 thru PS-667. The signal from the LS-RE09s is longest so these valves must operate at the time the LS-RE09s operate. Time required for actuation is one hour, and for isolation is 30 days. See Item 3.
13.5	SV-4916 4899 4897 4892		These are manual isolation valve control solenoid valves. It is assumed they will be closed by the operator at the same time as the auto valves above.
14.	SV-9151 9152 9153 9154	30 Days	These valves are needed to operate the vent valves which provide a path to relieve vacuum in the containment. Vacuum relief may be needed if the break assumed is small enough that containment isolation due to high containment pressure does not occur. Under these conditions, all the containment air is displaced by steam and subsequently condensed, thus creating a vacuum. The time required, therefore, is 12 hours for operation. This time is based on one hour for low reactor water level to be reached and a reasonable time to spray containment and condense the steam. Figure 13.1 of the FHSR indicates that it will take less than ten hours to depressurize the enclosure to atmospheric pressure following a large LOCA. However, they must not operate after vacuum is relieved due to containment integrity considerations. Time needed is 30 days.
15.	LT-3180 3181 3182 3183	30 Days	These transmitters provide level indication and control for the RDS system. After the RDS has operated, the reactor water level transmitters are used for indication only. Time needed is 30 days based on TMI lessons learned.
	3184 3185 3186 3187	1 Hour	LT-3184 through LT-3187 are the steam drum level transmitters which are only needed to fire the RDS. The time needed for these is one hour.

16.	SV-4980 4981 4982 4983 4984 4985 4986 4987	30 Days	These valves operate to open the RDS isolation and depressurizing valves. If a break occurred that was such a size that the primary system could refill, and repressurize subsequent to the first RDS actuation, it may be necessary to manually actuate the RDS. Time required is 30 days.
17.	Containment Penetrations	30 Days	These are needed for containment integrity.
18.	TB-240 241 242 243	30 Days	RDS system terminal boxes. See Items 15 and 16.
19.	LS-RE06A B RE20A B	10 Min	These level switches provide a scram signal for low steam drum level. The operator will manually scram the reactor for any loss of inventory event (EMP 3.3). Ten minutes is considered to be a reasonable time for action to trip the reactor, thus the time needed is ten minutes. The LS-RE20s also provide drum level indication to the operator in the control room. Steam drum level can also be obtained from the RDS level transmitter.
20.	PS-664 665 666 667	1 Hour	These pressure switches are used to provide a containment high-pressure scram and isolation signal. The longest time needed would be for a small break which does not cause automatic reactor scram and vent valve closure. The operator is instructed to manually trip the reactor and close the containment vent valves on indications of a small LOCA or steam line break. Vent valve closure will cause the containment to pressurize and the isolation scram to occur. A time required of one hour allows adequate time for manual scram and containment isolation thereafter.
21.	SV-NC22A B	10 Min	These solenoids provide a backup vent path for the scram inlet and outlet valves. They will operate (de-energize) on the first scram signal. The time required is 10 minutes. See Item 19.
22.	SV-NC22F G H J	30 Days	These valves are used to unisolate the scram dump tank by de-energizing one minute after a scram in order to equalize pressure between the tank and the reactor vessel to prevent control rods from drifting out of the core. The large LOCA will provide the worst conditions with regard

			to temperature-pressure. The first scram, probably low steam drum level, will occur very quickly. The valve must stay energized for one minute after the scram and then remain in the de-energized position for 30 days.
23.	PT-IA07C	30 Days	This pressure transmitter provides signal for reactor pressure indication. TMI lessons learned requires this instrument for 30 days.
24.	PDIS-7814	21 Hours	This differential pressure gage/switch measures the differential pressure across the basket strainer in the fire line cooling the core spray heat exchanger. It also provides an alarm in the control room for high differential pressure. It is only needed (see procedure SOP-8) when feeding core spray water via MO-7072. Therefore, it is needed until recirculation occurs. The environment will be normal until recirculation begins and its usefulness ends.
25.	Core Spray Pumps #1 #2	30 Days	These pumps provide core spray in the recirculation mode. They are needed for 30 days.
26.	Panel C30 Terminal Blocks	1 Hour	Required until the functions of LS-RE06, LS-RE20 and PS-IG11 are completed. PS-IG11 is the longest needed at one hour.
27.	SV-NC27 A5A Through F5B	10 Min	These solenoid valves control the scram inlet and outlet valves. These are needed to de-energize on the first scram.
28.	JB-160 161 164 166 167 180 170 170A 170B 171 171A 171B IG11A IG11B IG11C IG11D	1 Hour	These junction boxes serve instruments LS-RE09A - H, PS-RE15A - D, and PS-IG11E - H. The junction boxes do not have to last longer than the instruments. JB-IG11A - D serve instruments PS-IG11A - D and must last as long as LS-RE09A - H.

29.	Core Damage Monitor	30 Days	This monitor is used to detect radiation dose rates through the containment ·ˀll in the electrical penetration room so ˈt an estimate of core damage can be made.

E. Environmental Parameters for Hostile Areas

I. Containment

a.	Temperature	See Figures 1, 3		
b.	Pressure	See Figures 2, 3		
c.	Chemical	None		
d.	Humidity	100%		
e.	Thermal Aging	40 Years + Function Time		
f.	Submergence	Below Elevation 590'		
g.	Radiation - Rads - 30 Days	2 Hours	24 Hours	30 Days
	Gamma - Air	2.0×10^5	4.9×10^5	7.3×10^5
	Gamma - Water	1.0×10^6	1.1×10^6	2.8×10^6
	Beta - Air	1.3×10^6	4.2×10^6	1.3×10^7
	Beta - Water	4.0×10^5	7.3×10^5	1.2×10^6

References

1. Letter to Division of Reactor Licensing, R B Sewell to NRC dated May 15, 1975.

2. Letter to Director of Nuclear Reactor Regulation, D A Bixel to D K Davis dated February 24, 1978.

3. Letter from J L Beer to R W Sinderman dated October 15, 1980, 30-Day Integrated Radiation Doses for Electrical Equipment Qualification - Big Rock Point.

II. Pipe Tunnel

		HELB	LOCA
a.	Pressure	2.2 Psig*	0 Psig
b.	Temperature	210°F*	Ambient
c.	Chemical	None	None
d.	Humidity	100%	20% - 80%
e.	Thermal Aging	40 Years + HELB Event	40 Years
f.	Submergence	None	None
g.	Radiation Doses - Rads - 30 Days		
	Gamma	4.9×10^3	Sphere Surface 7.6×10^4
			6' From Sphere 5.2×10^4
	Beta	7.0×10^3	None

*According to Reference 3, the low reactor water level scram (containment isolation signal) occurs after 46 seconds. The main steam isolation valve requires 60 seconds to close after the low reactor water level scram set point. According to Reference 4, the pressure peaks at 2.2 psig in 1.5 seconds. To summarize, the peak temperature and pressure occurs in 1.5 seconds and is maintained for approximately 2 minutes before the steam source is isolated.

References

1. Letter from S A Giusti, Bechtel, to T A Bjornard, Consumers Power, dated March 8, 1973.

2. Letter from J L Beer to R W Sinderman dated October 15, 1980, 30-Day Integrated Radiation Doses for Electrical Equipment Qualification - Big Rock Point.

3. Computer code Safe C/BWR1 dated 6/6/75, run 5 S228 Case 2, 0.6303 ft^2 break.

4. Letter from G J Walke, Consumers Power, to J F O'Leary, NRC, dated June 29, 1973.

III. Electrical Penetration Room

	HELB	LOCA
a. Pressure	0 Psig	0 Psig
b. Temperature	40° - 100°F	182°F*
c. Chemical	None	None
d. Humidity	20% - 100%	20% - 100%
e. Thermal Aging	40 Years + HELB Event	40 Years + LOCA Event
f. Submergence	None	None
g. Radiation - Rads - 30 Days		
Gamma	Negligible	Sphere Surface 7.6×10^4
		6' From Sphere 5.2×10^4
Beta	Negligible	None

*This temperature is taken from data in Reference 3. For time of duration, the temperature envelope in Figure 3 conservatively is used (ie, draw a horizontal line on Figure 3 at 182°F).

References

1. Letter from J L Beer to R W Sinderman dated October 15, 1980, 30-Day Integrated Radiation Doses for Electrical Equipment Qualification - Big Rock Point.

2. Letter from K R Goller to R B Sewell dated November 5, 1974, Safety Evaluation Report on pipe breaks outside containment.

3. CONTEMPT run BILL030, 10/16/80, 0.63 ft^2 steam line break with decay heat.

IV. Sphere Ventilating Room

		HELB	LOCA
a.	Pressure	0 Psig	0 Psig
b.	Temperature	40° - 90°F	40° - 90°F
c.	Chemical	None	None
d.	Humidity	20% - 100%	20% - 100%
e.	Thermal Aging	40 Years	40 Years
f.	Submergence	None	None
g.	Radiation - Rads - 30 Days		
	Gamma	Negligible	Containment Surface 1.9×10^5
	Beta	None	None

References

1. Letter from J L Beer to R W Sinderman dated October 15, 1980, 30-Day Integrated Radiation Doses for Electrical Equipment Qualification - Big Rock Point.

2. Temperature - Reactor Building Temperature Log Sheet data for supply air taken from January through August 1978.

V. Core Spray Room

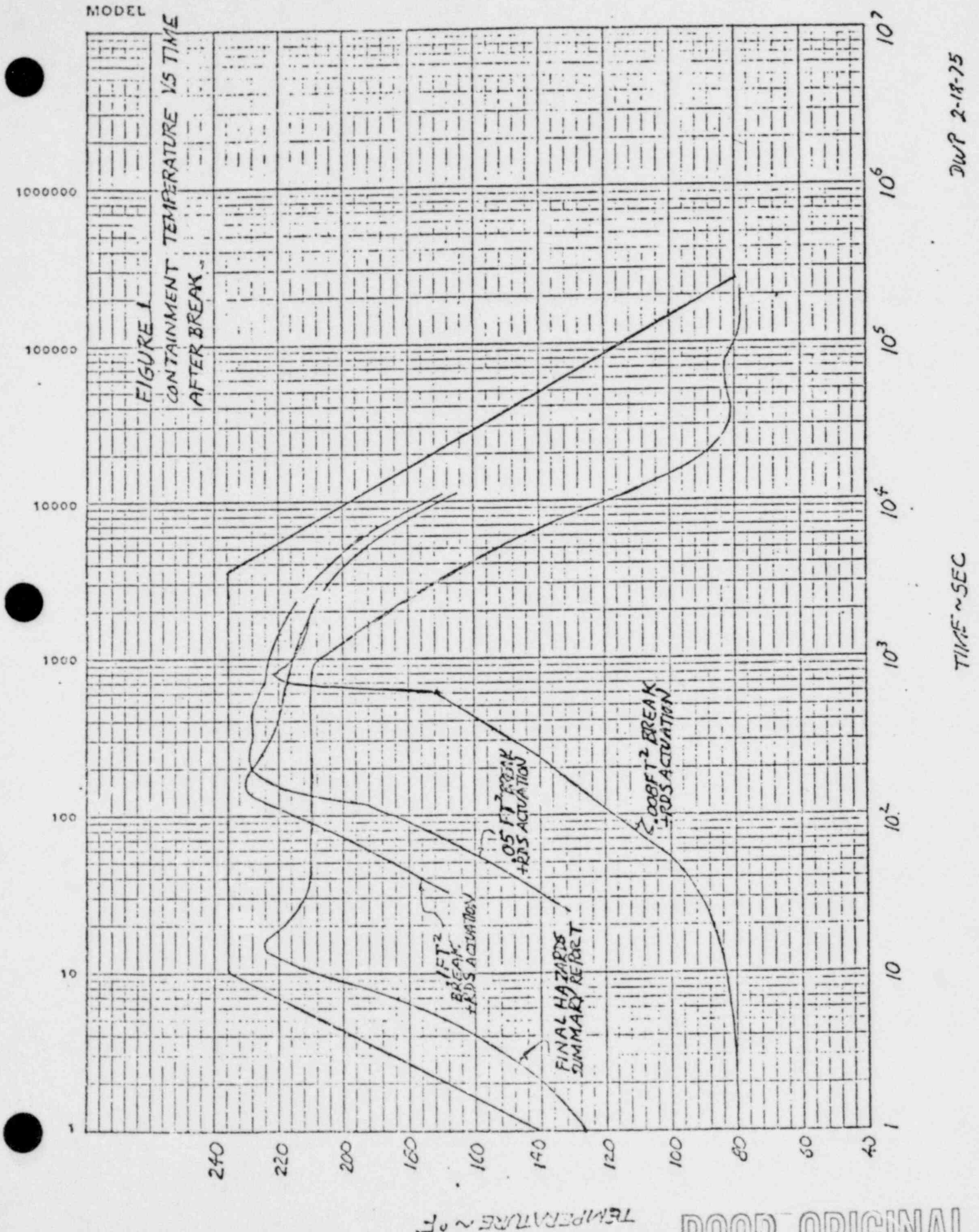
		HELB	LOCA
a.	Pressure	0 Psig	0 Psig
b.	Temperature	60° - 90°F	152°F*
c.	Chemical	None	None
d.	Humidity	20% - 80%	20% - 80%
e.	Thermal Aging	40 Years	40 Years + LOCA Event
f.	Submergence	None	None
g.	Radiation - Rads - 30 Days		
	Gamma	Negligible	4.0×10^4
	Beta	None	None

*This is the peak temperature reached in the core spray room 400 hours after the start of the recirculation mode. This temperature is constant for the remainder of the 30 days.

References

1. Letter from J L Beer to R W Sinderman dated October 15, 1980, 30-Day Integrated Radiation Doses for Electrical Equipment Qualification - Big Rock Point.

2. Bechtel Calculation #12447-053-M-1, October 28, 1980.


MODEL
FIGURE 1
CONTAINMENT TEMPERATURE VS TIME AFTER BREAK
1FT² BREAK +RDS ACTUATION
.05 FT² BREAK +RDS ACTUATION
.008FT² BREAK +RDS ACTUATION
FINAL HAZARDS SUMMARY REPORT
1000000
100000
10000
1000
100
10
1
TIME ~ SEC
TEMPERATURE ~ °F
240
220
200
180
160
140
120
100
80
60
40
DWP 2-18-75

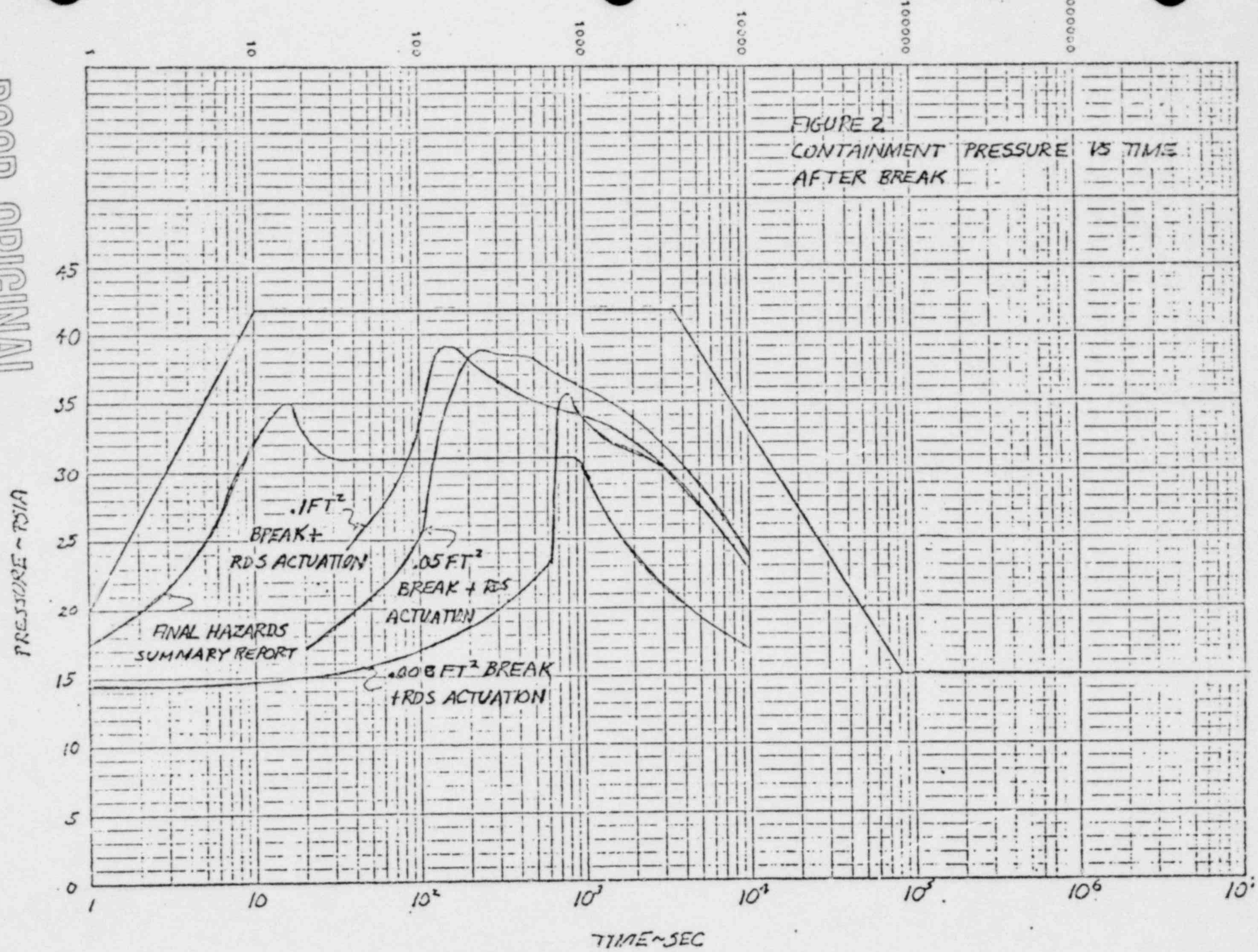


FIGURE 2
CONTAINMENT PRESSURE VS TIME
AFTER BREAK
.1FT²
BREAK +
RDS ACTUATION
.05 FT²
BREAK + RDS
ACTUATION
FINAL HAZARDS
SUMMARY REPORT
.008 FT² BREAK
+RDS ACTUATION
PRESSURE ~ PSIA
TIME ~ SEC

DWP 2-18-75

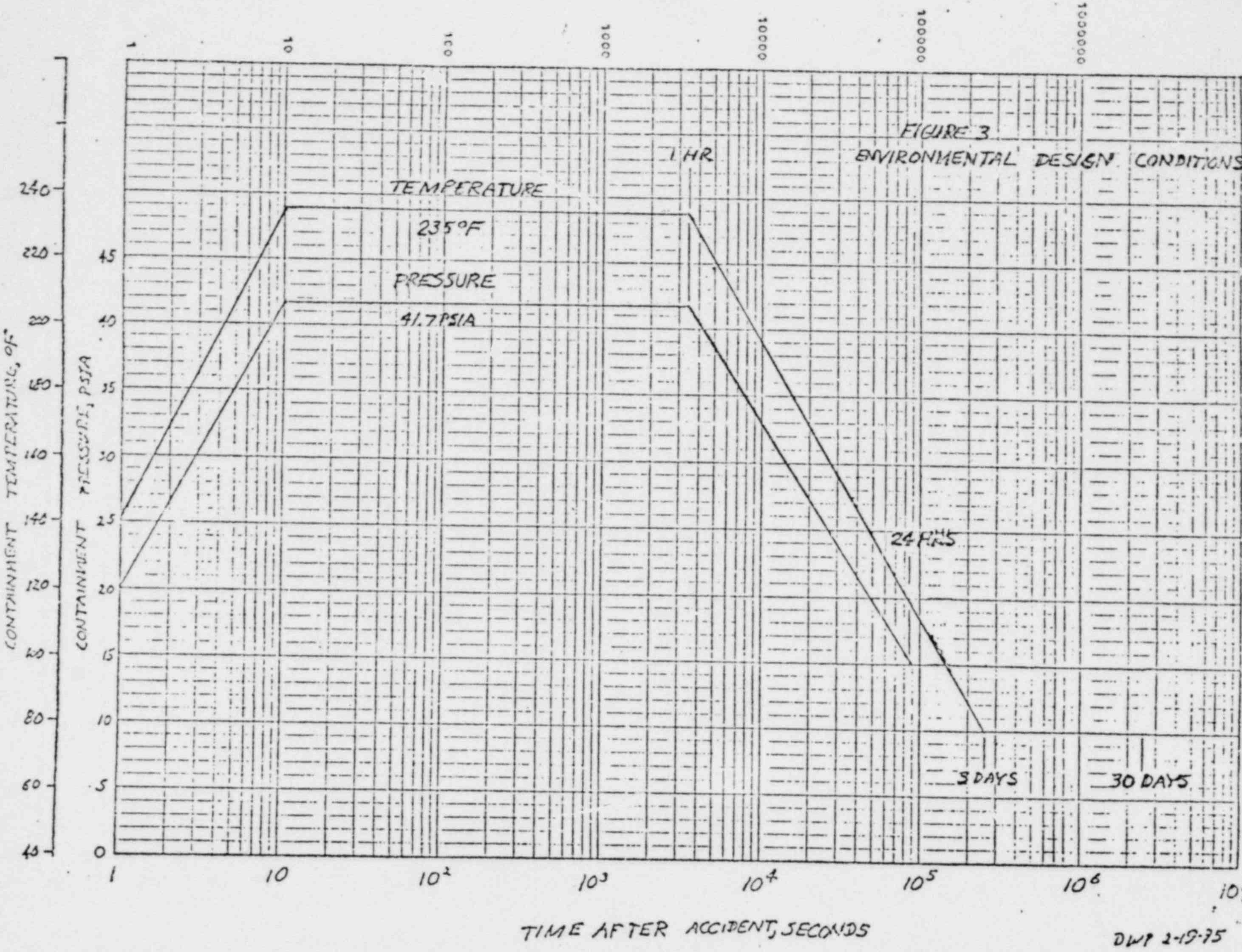


FIGURE 3
ENVIRONMENTAL DESIGN CONDITIONS
TEMPERATURE
235°F
PRESSURE
41.7 PSIA
1 HR
24 HRS
3 DAYS
30 DAYS
CONTAINMENT TEMPERATURE, °F
CONTAINMENT PRESSURE, PSIA
TIME AFTER ACCIDENT, SECONDS
1
10
100
1000
10000
100000
1000000
10
10^2
10^3
10^4
10^5
10^6
10^7
240
220
200
180
160
140
120
100
80
60
40
45
40
35
30
25
20
15
10
5
0
DWP 2-19-75

SECTION III

RADIATION INSIDE AND OUTSIDE CONTAINMENT

The source terms used to calculate containment doses and all subsequent doses came from the ORIGEN code. The gamma and beta energies per disintegration for each nuclide are used. The beta energies are from the ORNL MEDLIST and the gamma energies were compiled from a listing for the computer program RASTUS. The RASTUS code calculates doses using Equation II as found in a letter from D A Bixel to J G Keppler dated May 11, 1977 regarding Consumers Power Company's response to IE Report 050-155/77-04.

Containment

Containment beta and gamma dose rates for air and liquid were calculated at t=0, 2 hours, 24 hours and 720 hours. Infinite cloud calculations were performed to obtain dose rates in all cases except the gamma dose rate in air.

The 30-day integrated radiation doses were obtained by segmenting time periods and summing the products of each time period multiplied by the average logarithmic dose rate in that time period. The results are as follows:

Containment Air Beta	1.32E+7 Rads
Containment Air Gamma	7.3E+5 Rads
Containment Water Beta	1.15E+6 Rads
Containment Water Gamma	2.83E+6 Rads

Beta doses were applied to cable only as the equipment provides adequate shielding for beta radiation.

Electrical Penetration Room

The gamma dose rate due to containment shine in the electrical penetration room has been previously calculated. The methodology used was the same as referenced above (D A Bixel to J G Keppler, May 11, 1977). As was shown in that submittal, the electrical penetration room receives radiation from only 35% of the containment as a result of its location relative to the internal structures. Credit is also taken for shielding from the containment shell. The 30-day integrated dose was obtained by the same method described under containment doses. The shine dose at six feet from the containment surface is 5.21E+4 rads. The dose rates at six feet and on the containment surface were ratioed and multiplied by the 30-day integrated dose at six feet to obtain the 30-day integrated dose at the containment surface. The dose at the surface was found to be 7.6E+4 rads.

Sphere Ventilating Room

The radiation dose due to gamma shine at the air shed was calculated by using the following three volumes of activity. One is in the vicinity of the cooling unit, another in front of the clean-up demineralizer and a third as the top half of the containment sphere. Using the whole top half of the containment sphere is conservative si the shielding around the steam drum would diminish the dose rate. The cor uter program RASTUS was used to calculate the dose rates from three equivalent volume spheres. The two smaller spheres were taken to be of equal volumes and distances to the air shed. The radius of the smaller spheres is 3.81E+2 cm with a distance of 3.45E+2 cm to the air shed. The larger sphere has a radius of 1.47E+3 cm with a distance of 2.07E+3 cm to the air shed. Shielding used for all three shields is 3/4" of iron.

The dose rates from the three spheres as provided by RASTUS (runs 478 through 485) were added for each time period. The integrated dose is 1.94E+5 rads.

Core Spray Room

The gamma radiation dose due to recirculating fluids in the core spray room was calculated. The calculation involved modeling the piping carrying radioactive sources in the room by using spheres of diameters slightly larger than the pipe diameter to account for all of the volume. The dose in the room is then a sum of the contributions from all of these sources. This calculation gives a 30-day integrated dose of 2.64E+3 rads; however, it assumed a 10% core melt and no noble gases or particulates in the liquid phase. Regulatory Guide 1.3 calls for 100% of the noble gases and 1% of the particulates in the liquid phase.

To update this dose to the Regulatory Guide 1.3 source terms, it was first multipled by ten for the 100% fuel melt case to give 2.64E+4 rads. To add the contribution from noble gases and particulates, a recent RASTUS run at t=2 hours for containment air was used to compare the contribution from noble gases and particulates vs halogens. The total dose rate due to noble gases and particulates was found to be roughly half of the total dose rate due to halogens (halogen dose rates were multiplied by 50 since the RASTUS run was for 1% of the halogens at t=2 hours and the liquid phase contains 50% of the halogens).

The total 30-day integrated dose is therefore:

$$(2.64E+4) + (2.64E+4)(0.5) = 3.96E+4 \text{ Rads}$$

This is conservative since the dose rate ratio of noble gas and particulates to halogens becomes less at times greater than two hours.

Doses from the containment atmosphere were not considered in this calculation due to the heavy shielding around the core spray room.

Pipe Tunnel

Beta and gamma 30-day integrated doses were calculated for the turbine building given the following initial assumptions:

A. The entire primary coolant volume (3,830 ft^3 from Technical Specification 4.1.2) is dumped to the turbine building and it all flashes to steam.

B. The main steam isolation valve closes before any fuel failure occurs and no fuel failure occurs after closure.

SECTION IV

AGING

Most areas outside containment are maintained in a suitable environmental condition by HVAC equipment. The normal maximum environmental design temperature is 100°F which in this study is defined as a "nonharsh environment." Thermal aging for these components is not required if the normal and post-accident environments are nonharsh. Equipment designed and installed per industry standards would be capable of satisfactory operation without exhibiting age-related degradation due to temperature up to 104°F. Therefore, the effects of thermal aging are not considered for components located in areas where the maximum ambient temperature can be maintained at 104°F or below.

There is a class of equipment for which no preaging tests were performed and for which a documented conclusion of insensitivity to aging cannot be reached. There are, in general, two reasons for this. The first is inability to qualify due to lack of precise knowledge of the materials of construction or inability to determine the exact aging characteristics of the known materials. Lumped in this category are: Uncooperative manufacturers; cases where no manufacturing records were kept; imprecise definitions of materials such as wax, lacquer, cardboard, plastic, resin, fiber glass laminate, etc; as well as cases where exotic trade name materials were used and cannot be traced to more generic names; or where no aging data can be found. The second reason relates to equipment with known materials of construction where at least some aging data are available, but where the existing data would suggest that the equipment is age sensitive. The bounds of this category range from equipment for which aging calculations suggest the equipment has a 40-year life but the equipment was not aged prior to LOCA testing to equipment for which Arrhenius calculations indicate a life of as low as two years.

The most prevalent equipment in the unknown category is equipment with electrical windings (motors, solenoids) and general use electrical components (wire nuts, splices, tape). The most prevalent equipment in the short age category are seal systems (O-rings, gaskets), bellows and diaphragms (solenoid valves, pressure transmitters).

In general, the equipment will have to be qualified by test shown to have materials whose lifetime is acceptable or be replaced with qualified equipment prior to the deadline date of June 30, 1982; or, in the case where parts qualifiable for long life are not available, be placed on a rotating replacement schedule.

In all cases, the question of justification for continued operation while the final remedy is implemented arises. The following generally applicable arguments are presented here to avoid excessive duplication of wording in the equipment list. When they are available, other equipment specific arguments and temporary remedies are presented in the equipment list.

The three arguments presented are closely interrelated and, while each has its own merits, they are more compelling when taken together.

Aging Models and Technology - "Aging" is a very complex yet universal phenomenon. The number of variables to be considered is very large, perhaps infinite, and a satisfying scientific approach that can be defended against all valid questions is clearly beyond the capability of today's state of knowledge. What is really desired in the present instance is a regression of equipment reliability during LOCA as a function of age. Each piece of equipment is constructed of various materials from parts made by various processes; has its unique demands on various characteristics of its materials of construction; fails to perform when varying numbers of material failures occur; and, is exposed to its own unique history of environments and duty cycle in reaching a given chronological age at which various LOCA (or other accident or transient conditions) are postulated to occur. To produce the data for direct regression is clearly impossible so a very long layered process of assumption and approximation is employed. In most instances, these assumptions are imbedded in both the experiments from which the data are taken and the analysis of the data (regression). For instance, if temperature is chosen as the relevant environmental parameter, the equipment (material) is exposed to heat. The failure data (material characteristics) are regressed against temperature and time, and the time variable is equated to aging. In this process of severe simplification and layered approximation, conservatisms tend to creep in at every approximation in each layer (most of the experimental inaccuracy, lack of experimental control and data scatter go toward failure - the objective being to demonstrate what it can withstand vice, what conditions will positively fail it). An excellent discussion of the variety of modeling techniques, as well as a good example of the varying amount and quality of existing data, are provided in EPRI Report NP-1558.

In arriving at an aging model, the following approximations are usually made:

a. Temperature is assumed to be the variable that controls aging.

b. Materials defined by trade name or major chemical constituent are assumed the same, at least when quoting references to existing test data, and particularly when the results tend to be negative.

c. Geometry of the part is given no special significance. When the data result from individual material tests, the samples are usually very thin (typically a few mils).

d. The way the material is exposed (direct or protected) is given no particular significance. In most tests, exposure is direct even though in practice the material is usually partially protected (air circulation, etc).

e. Tests (especially isolated material tests) tend to equate a somewhat arbitrary reduction in a particular material characteristic (tensile strength, percent elongation, etc) to failure. This characteristic may or may not be important to any specific application of the material. In any event, it is usually chosen very conservatively (sometimes as little as 25% reduction).

f. The Arrhenius equation is assumed to apply and to accurately relate damage at one temperature to damage at any other temperature. The basis for the Arrhenius equation appears to be as much or more empirical than it is theoretical. Judgments as to whether it should be applied in any given situation are very difficult to make. In most cases, its application leads to conservative results.

Due to the state of development of accelerated aging as a science, the models tend to be conservative. Very little comparison data between model predictions and actually aged power plant equipment exist. Therefore, it is difficult to improve models and remove conservatisms. The equipment to monitor actual in-service environment and equipment duty cycle would be extensive. Existing programs such as Nuclear Plant Reliability Data System (NPRDS) have no provisions to collect even estimates of such environmental data.

Conservatism of Model Input - In order to apply the aging models, the operating environment of the installed equipment must be known. Since instrumentation to measure and record the history of actual equipment operating temperatures does not exist, it is, at best, possible only to establish a range of operating temperatures. In most cases, the maximum specified operating temperature of the equipment is used in the Arrhenius equation. Actual operating temperatures, both peak and average, are usually lower. Since the Arrhenius equation is exponential in temperature, this can be a significant conservatism.

In cases where exact Arrhenius equation constants for the material are unavailable or in cases where the literature contain more than one value, the most conservative available constants are used. Again, due to the exponential nature of the relationship, significant conservatisms can be introduced (activation energies from 0.5 to 1.5 encompass six orders of magnitude in the aging rate).

The objective, in general, was to find the components insensitive to aging rather than simply having the required life. This is necessary to justify not preaging equipment before LOCA testing. Usually the required lifetime to make an insensitivity finding is of the order of 400 years.

Plant Experience Vice Model Output - In some cases, like "BUNA N" in solenoid valves, the models predict very short lives. There are, however, instances where the equipment (valves) has been installed, has operated and has passed periodic surveillance tests for intervals as long as 10 to 15 years without significant failure rates. The numbers of such pieces of equipment (valves) are sometimes in the hundreds. Explanations for this are generally similar to those given above in discussions of the conservatisms of the model and the input. Their existence, however, would seem to be reason enough to place significant credence in the above discussion.

The situation in the plant is that most equipment is in a cool environment which oscillates seasonally ranging from 40°F or 50°F in the winter to 90°F or 100°F in summer. Under these conditions, significant aging is not thought to have occurred. This is further proved in the case of BUNA N. BUNA N has an activation energy of 0.56. Most materials have activation energies of from 0.5 to 1.5 with most clustered around 1.0. The activation energy is taken from a test in which the property most sensitive to aging was found to be elongation. Failure was defined as a 20% decrease in elongation. The properties which are important in most cases in instruments are compression set, hardness and tensile strength. The lifetimes for these properties are greater, but given an activation energy and constant for BUNA N at 20% decrease in elongation, the life at 70°F is 11 years. Failure will occur at times greater than this.

Since the plants frequently contain a number of similar pieces of equipment, and since one expects significant statistical spread in failure rates, it would seem that a lack of observed failures would indicate that most of the equipment has a good portion of its usable life remaining. The statistical spread is generally acknowledged (Chapter 8 of EPRI Report NP-1558). It would also be expected, due to different operating environments (rooms, cooling air flows, duty cycle) as well as normal spread in material characteristics (batches, slight inhomogenities) plus actual state of stress in the as-assembled component (stretch in the diaphragm, pinch of the gasket, draw in the wire insulation). Monitoring of failure rates to detect the absence of the statistical tail should add additional statistically quantifiable assurance that the actual plant equipment is not seriously degrading due to aging to the point where common-mode failure is a significant concern - at least for a relatively short period of time until satisfactory replacement components could be obtained.

SECTION V

ELECTRICAL CABLE

Bechtel Corporation was contracted to provide the necessary information to show the qualification of electrical cable in the circuits containing the electrical equipment shown on the equipment list in Section II.C. The general methods used to ascertain the materials of construction of the cables are described in this section.

A walk down of the electrical cable systems at the Big Rock Point was carried out from 10/13 to 10/17/80. The purpose of the walk down was as follows:

a. Inspect safety-related cabling located in hostile areas, including pigtails on the containment penetrations.

b. To the extent possible, identify the manufacturer for each type of cable.

The following procedure was followed to accomplish the walk down objectives:

a. By referring to the Plant Circuit Schedule, the type of cable, cable code and cable routing were established for all cable runs associated with devices shown on the Equipment List, Section II.C.

The Electrical Penetration Index was used to identify the location of penetrations and the associated pigtails.

b. Based on cable routing, a convenient and accessible location was selected for each cable run. The cable was visually inspected at this location with the help of a bright beam flashlight. Every attempt was made to identify manufacturer's markings on the conductors and jackets. Further, conductor insulation and jacket were carefully examined to detect possible signs of degradation such as cracks, brittleness, loss of flexibility, discoloration, etc.

c. If a particular type of cable was used, both inside and outside containment, the cable was inspected at the more hostile location inside the containment since degradation was more likely to occur at the hostile location.

d. The outer diameters over the conductor insulation of selected cables of different types were measured with a micrometer to verify the observations against the data shown on the Purchase Orders.

The following conclusions are reached, based on the preparatory data gathered by document review, and on the actual data gathered by walk down:

a. During 1961 and 1962, Purchase Orders for various types of cables were issued on Rome Cable Corp, Electric Wholesale and Supply Co and Fitzpatrick Electric Supply Co.

b. The above cables can be divided into the following three main categories based on the types of insulation and jacket material:

   i. Polyethylene (PE) insulation with polyvinyl chloride (PVC) jacket.

   ii. Polyvinyl chloride (PVC) insulation with polyvinyl chloride (PVC) jacket.

   iii. Butyl rubber (BR) insulation with polyvinyl chloride (PVC) jacket.

c. Visual inspection during walk down indicated that there were no identifying marks on the insulation or jacket which would enable identification of the cable manufacturer for the cables ordered during 1961 and 1962.

d. Visual examination indicated that there were, in fact, three (3) different types of 1961-62 vintage cable installed in the field, based on the physical appearance of various cables. For example, the Purchase Orders and circuit schedule indicated that certain cable runs used PE/PVC cables and other cable runs used BR/PVC cables. Visual inspection revealed that each type of cable appeared to be unique; that is, based on physical appearance, including texture and finish of insulation material, BR/PVC cables can be clearly distinguished from PE/PVC cables.

e. Measurements of outer diameters over the conductor insulation of each type of cable with a micrometer have confirmed that the dimensions of installed cables are closely in agreement with the dimensions shown on Purchase Orders.

f. The field verification that, (i) there are, in fact, three types of 1961-1962 vintage cables installed in the plant and, (ii) the cable dimensions agree closely with the dimensions shown on the Purchase Orders, lends a considerable degree of credibility to authenticate the technical details shown on the Purchase Orders. It is, therefore, concluded that, even though these cables lack the manufacturer's identifying marks, the insulation/jacket materials are PE/PVC, BR/PVC or PVC/PVC, as shown on the circuit schedule and Purchase Orders.

g. These cables, which were installed in 1961-1962, have remained in service for approximately 19 years. A very careful visual examination of several cables of each type at various locations, including hostile locations, indicates that both conductor insulation and jackets appear to be in excellent condition and there are absolutely no signs of degradation, such as cracks, brittleness, loss of flexibility, discoloration, etc.

LIST OF DEVICES WITH SCHEME NUMBERS, CABLE NUMBERS AND TYPES OF CABLES

LIST OF DEVICES WITH SCHEME NUMBERS, CABLE NUMBERS AND TYPES OF CABLES

Device Identification	Cable No	Cable Type	Scheme No	
SV-4922	C02-Z04/6	5	6505	
PS-664	C02-Z10/6	5	6514	
PS-665	C02-Z11/7	5	6514	
PS-666	C02-Z10/7	5	6515	
PS-667	C02-Z11/8	5	6515	
SV-NC22A	C02-Z46/1	5	6514	
SV-NC22B	C02-Z45/2	5	6515	
SV-NC22F) SV-NC22G)	C02-Z50/2	5	6514	
SV-NC22H) SV-NC22J)	C02-Z50/2A	5	6515	
Core Spray Pump 1	P12-R01/9	25	5415	
SV-9153) and) SV-9154)	C02-C26/1		8512	(Rockbestos Fire Wall Type TC per Field Verification)
SV-4891	C01-Z05/4	5	7501	
SV-4869	C01-Z03/6	5	7501	
SV-4879	C01-C13/1	5	6505	
SV-4876	C15-Z02/1 C02-Y02/2	5 5	6505 6505	
MO-7050LS	C02-R01/1	5	6605	
MO-7050LS	D01-R01/14	5	6605	
MO-7050 (Motor)	D01-R01/15	24	6605	
PT-IA07C	C02-Z09/6	53	6803	

Device Identification	Cable No	Cable Type	Scheme No
SV-4896	C01-Z04/7	5	7501
SV-9151	C01-C26/1	4	8501
SV-9152	C01-C26/2	5	8501
LT-3171	ES3171-LT3171/1	214	LI-3326
LS-RE09E	JB160-LS/RE09E/1	112	B163
LS-RE09F	JB160-LS/RE09F/1	112	B152
LS-RE09G	JB160-LS/RE09G/1	112	B163
LS-RE09H	JB160-LS/RE09H/1	112	B152
MO-7064	R01-R01/3	EPR 2/C #12	5414
MO-7068	N01-R01/33	21	5414
	N01-R01/3	22	5414
	/34	214	5414
	/35	214	5414
MO-7070	M(H-76)-N63/2	714	B163
	M(H-76)-N63/3	214	B163
	M(H 76)-R63/1	112	B163
	M(H-76)-R63/2	514	B163
SV-4892	C01-Z01/9	5	7501
SV-NC27A2A	C02-Z14/5	5	6514
LS-3565	C01-Z01/5	21	6503
LS-3564	C01-Z02/5	21	6503
LS-3563	C01-Z03/3	21	6503
LS-3562	C01-Z04/4	21	6503
LS/RE09C	C02-Z02/18	5	6514
FT-2161	ES2161-FT-2161/1	214	FI2326
FT-2162	ES2162-FT-2162/1	214	FI2327

Device Identification	Cable No	Cable Type	Scheme No
FT-2163	ES2163-FT-2163/1	214	FI2328
FT-2164	ES2164-FT-2164/1	214	FI2325
LS/RE09D	C02-Z02/19	5	6515
PT-173	C02-Z05/17	XLPE Flametrol	8501
PT-187	C02-Z02/27	XLPE Flametrol	8512
PT-174	C02-Z01/88	XLPE Flametro	8512
LS-RE09B	C02-Z01/70	5	6515
LS-RE09A	C02-Z01/69	5	6514
MO 7061 (Motor)	D01-R01/5	29	5602
MO 7061 LS	D01-R01/4	30	5602
MO 7051 (Motor)	D01-R01/7	29	5601
MO 7051 LS	D01-R01/6	30	5601
MO 7066 LS	N01-R01/8	3	5418
MO 7066 Motor	N01-R01/9	22	5418
LS/RE09B	D01-Z01/1	21	5602
LS/RE09A	D01-Z01/2	21	5601
LS-3562) LS-3563)	Z03-Z04/5	21	6503
PS-IG11B	Z03-Z04/2	21	5601
PS-IG11B	Z03-Z04/3	21	5602
LSRE09C	Z01-Z03/3	21	5601
LSRE09D	Z01-Z03/4	21	5602

Device Identification	Cable No	Cable Type	Scheme No
SV-4916	Y01-Z06/1	1	9502
	Z05-Z06/4	1	9502
	Z06-Z07/5	1	9502
MO 7071	P22-N56/1	Type Not in Circuit Schedule (3/C #12)	B152
MO-7070	P22-N63/1	Type Not in Circuit Schedule (3/C #12)	B163
MO-7068	P22-N01/4	Type Not in Circuit Schedule (3/C #12)	5414
PS/1G11A)	Z01-Z02/5	21	5601
PS/IG11A)	Z01-Z02/6	21	5602
PS/IG11B	Z02-Z04/1	21	5601
PS-IG11E	JB160-PS/IG11E/1	112	B163
PS-IG11E	LS/RE09E-PS/IG11E/1	112	B163
PS-IG11F	JB160-PS/IG11F/1	112	B152
PS-IG11F	LS/RE09F-PS/IG11F/1	112	B152
PS-IG11G	JB160-PS/IG11G/1	112	B163
PS-IG11G	LS/RE09G-PS/IG11G/1	112	B163
PS-IG11H	JB160-PS/IG11H/1	112	B152
PS-IG11H	LS/RE09H-PS/IG11H/1	112	B152
JB170	JB170-LS/RE09A/1	214	5601
JB170	JB170-LS/RE09C/1	214	5601
JB171	JB171-LS/RE09B/1	214	5602
JB171	JB171-LS/RE09D/1	214	5602
SV-4899	C01-Z10/2	5	2502
SV-4897	C01-Z03/9	5	6505

Device Identification	Cable No	Cable Type	Scheme No	
SV-NC27A5A	C02-Z17/6	5	6514)	Typical
SV-NC27A5B	C02-Z22/2	5	6514)	All Other
SV-NC27A5C	C02-Z28/2	5	6514)	Cables Are
SV-NC27A5D	C02-Z34/1	5	6514)	Type 5
SV-NC27A5E	C02-Z40/2	5	6514)	to Other
SV-NC27A5F	C02-Z45/2	5	6514)	Solenoids
JB180	D01-Z03/1	21	5601 5602	
Core Spray Pump #2	P22-R01/10	25	3406	
PS-638	C02-Z31/1	5	701	
MO-7066 LS	N01-R01/8	3	5418	
MO-7066 (Motor)	N01-R01/9	22	5418	

SECTION VI

EQUIPMENT WORK SHEETS AND SUMMARIES

NOTE: Although it is stated that certain components will be replaced, Consumers Power Company reserves the right to demonstrate qualification by shielding, moving, determining that the components are not required for mitigation, demonstrating that the component has accepted failure modes or demonstrating minimal impact should failure occur.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous Electrical	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: Penetrations Types 1 through 7	Temperature (°F)	235	307	Section II.E	1, 2	Test & Analysis	
Component: Electrical Penetrations	Pressure (PSIA)	41.7	74.7	Section II.E	1, 2	Test & Analysis	
Manufacturer: General Electric	Relative Humidity (%)	100	100	Section II.E	1, 2	Test & Analysis	
Model Number:	Spray	Not Subject to Spray	-	-	-	-	
Purchase Order Number: 3159-E-19	Radiation (Rad)	1.373×10^7	2.5×10^8	Section II.E	Sheet 2	Analysis	
Function/Service: Transmit Electrical Signals/Power/Control	Aging	40 Yr + LOCA	40 Yr + LOCA	Section II.E	1, 2	Analysis	
Accuracy: Spec: Demo:	Submergence	Not Subject to Submergence	-	-	-	-	
Location: Containment							
Elevation: >590							
Flood Level Elevation 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Pittsburgh Testing Laboratory Report 1-C dated March 13, 1958 and Report 3-B dated April 17, 1958. 2. Specification No E-2345-104 for test of Typical Sphere Electrical Penetration Assembly Units prepared by Bechtel Corp, dated February 6, 1958.	1. Separate qualification reports are submitted herewith for the qualification of electrical wires used inside the penetration Types 1, 2, 4, 5, 6, & 7. Penetration Type 3 uses chromel and alumel wires for thermocouple circuits which are used in

45

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Letter from C A St Onge of Bechtel Power Corp to W Beckius of Consumers Power Company dated October 30, 1980. 4. "Effects of Radiation on Materials and Components" by Kircher & Bowman. 5. "Radiation Effects on Organic Material" by Bolt and Carroll.	non-Class 1E circuits. Therefore, no separate qualification report is submitted for these chromel and alumel wires. Penetration Type 6 uses bare copper rods for equipment ground. 2. Penetration Types 1 through 4 and Type 7: Rated for 600 volt circuits. Penetration Type 5: Rated for 3,000 volt circuit. Penetration Type 6: Used for equipment ground where soft drawn copper rods are passed through penetration.

Sheet 2

Component Electrical Penetrations (Types 1 Thru 7)

Type 1: 6" penetration assembly embedded with 20 feet long, 92 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 2: 6" penetration assembly embedded with 20 feet long, 52 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 3: 6" penetration assembly embedded with 20 feet long, 52 single conductor #14 AWG solid chromel & alumel wires with neoprene insulation firmly bonded to the conductor.

Type 4: 6" penetration assembly embedded with 20 feet long, 20 single conductor #8 AWG solid copper wires with GE versotol insulation and geoprene jacket.

Type 5: 6" penetration assembly embedded with 20 feet long, 6 single conductor #4/0 AWG 19 strand copper with Okonite oil-based rubber insulation and heavy-duty black neoprene jacket.

Type 6: 6" penetration assembly embedded with 5 feet long, 4 copper rods with 1/2" outer diameter and other insulating materials.

Type 7: 1-1/2" penetration assembly embedded with 20 feet long, 8 single conductor #14 AWG solid copper wires with GE versotol insulation and geoprene jacket.

2.0 The subject electrical penetrations are used for Class 1E as well as non-Class 1E circuits inside the containment. This qualification report, therefore, includes both the functional and the pressure boundary integrity of these penetrations.

2.1 Electrical Penetration Assemblies

Penetration Types 1 thru 6 consist of 6" nipples, 12" long with 6" conduit steel bushing at one end and a 6" to 8" concentric reducer welded at the other end. After fabrication of the complete penetration as per the procedure given below, the complete unit as such is welded to 8" Schedule 40 sleeve 12" long which in turn is welded to reactor building shell.

Penetration Type 7 consists of 1-1/2" nipple, 9" long with 1-1/4" nipple at one end and a 1-1/2" to 3" concentric reducer welded at the other end. After fabrication of the complete penetration as per the procedure given below, the complete unit as such is welded to 3" Schedule 40 sleeve which in turn is welded to lock shell.

Sheet 2 (Contd)

Fabrication Procedure: Fabrication of penetration units from 6" or 1-1/2" Schedule 40 nipples is performed in the following sequence:

a. Wire brush inside wall of the 12" long 6" nipple (or 1-1/2" nipple for Type 7) and swab with toluene solvent.

b. Thread the wire, cable or copper rod into the spacers and insert it into the nipple until Spacer Disc 1 reaches the bushing (or the end of the 1-1/4" nipple). Place the unit in the vertical position such that Spacer Disc 1 is resting on the bushing (or the end of 1-1/4" nipple) and is level.

c. Place a suitable length of thin wall conduit or other suitable tube through the center hole of Spacers 2 through 5 and insert tube to near bottom spacer. Pour a 2-3/4" to 3-1/8" layer of Chico "A" compound taking care not to splash any of the compound on the inside wall of the pipe above this layer. After completing the pour, withdraw the tube and move Spacer 2 into the nipple until small amount of Chico "A" begins to appear through various holes and outside edge of spacer. The Chico "A" is allowed to set for 3 days.

d. Remove all foreign matter and swab internal pipe surface with solvent. Pour 1/2" to 5/8" layer of epoxy resin (with glass bead filler) on top of the Chico "A" compound in the same manner that Chico "A" was poured. After completing the pour, withdraw the tube and move Spacer 3 into the nipple until small amounts of resin begin to appear through the various holes and outside edge of the spacer. The resin shall be allowed to set for at least 24 hours.

e. Pour 3" to 3-1/2" layer of Ozite "B" compound in the same manner as previous pours after first bringing the temperature of the Ozite to approximately 350°F. Spacer 4 is then moved into the nipple until small amounts of Ozite begin to appear through various holes and outside edge of the spacer. The Ozite "B" compound is allowed to set for at least 12 hours.

f. Pour 2-1/2" to 3-1/8" layer of Chico "A" compound in the same manner as previous pours. On 1-1/2" units only move Spacer 5 into the nipple until small amounts of Chico begin to appear through the various holes and outside edge of the spacer. Allow the Chico "A" to set for 3 days.

g. For 6" units only, pour a 1/2" to 5/8" layer of epoxy resin on top of Chico in the same manner as previous pours. Spacer 5 shall then be moved into the nipple until small amounts of resin begin to appear through the various holes and outside edge of the spacer. After the spacer is in position, pour approximately 1/4" float of resin over the top of spacer so that it is completely covered. The resin is allowed to set for at least 24 hours.

Sheet 2 (Contd)

h. For 6" units, a 16" extension nipple is also attached to the penetration unit to provide a raceway for carrying the conductors to the tray system and to permit, in combination with the fiber barrier, the addition of a thermoplastic compound to increase further the reliability of the penetration.

2.2 The qualification information of the subject electrical penetrations is based on the following documents and discussions:

2.2.1 Document Reference 1 includes Pittsburgh Testing Lab (PTL) Report 1-C dated March 13, 1958 and Report 3-B dated April 17, 1958. These test reports cover actual leakage rate test and LOCA tests on similar design penetrations made by General Electric. The similarity between electrical penetration Types 1 thru 7 and the penetration assembly specimens covered by PTL reports is based on the following facts:

a. Production procedures used for the assembly of both types of penetrations are same.

b. Both types of penetrations are designed by General Electric.

c. Similar insulating materials such as Textolite, Chico, epoxy and Ozite "B" have been used for the assembly of both types of penetrations.

d. Penetration Type 1 which consists of 92 single conductor #14 AWG solid copper is similar to 3 units of 92 single conductor #12 AWG solid copper tested by PTL.

e. Penetration Type 5 which consists of 6 single conductor #4/0 AWG stranded copper used in 3 kV circuits is similar to 2 units of 6 single conductor #4/0 AWG for 5 kV circuits tested by PTL.

f. Penetration Types 2, 3, 4 and 7 are similar to 92 single conductor #12 AWG solid copper tested by PTL.

During environmental qualification testing covered by PTL Reports 1-C and 3-B, the penetration assemblies were first subjected to air pressure of 60 psig (74.7 psia) for one hour and then subjected to steam exposure test at 307°F, 60 psig (74.7 psia) and 100% relative humidity for 23-1/2 hours. At the end of the test, the units were cooled to room temperature and successfully passed a 150 psig cold hydrostatic pressure test for one minute without any leakage.

2.2.2 Document Reference 2 includes Specification E-2345-104 for test of typical sphere electrical penetration assembly units prepared by Bechtel Corp dated February 6, 1958. This specification covers the qualification testing of electrical penetration units by Pittsburgh Testing Laboratory described in Subparagraph 2.2.1 above.

Sheet 2 (Contd)

2.2.3 Analysis of materials used in the fabrication of complete penetration, as covered by Document References 3 and 4, confirms the following:

a. That the properties of Chico "A" compound are similar to Portland Cement which, being inorganic in nature, is not subject to radiation aging.

b That the Textolite spacer discs used in the penetration assembly are similar to epoxy or phenol formaldehyde laminates with glass fabric and possess radiation and thermal withstand properties which are equal to or better than regular epoxy resins. Per Document Reference 4, this type of material can withstand up to 8.3×10^7 R as given in Table 3.22 on Page 156.

c. That the epoxy compound used in the penetration assembly is good for maximum continuous service temperatures up to 400°F.

2.2.4 Document Reference 4 on Page 96 indicates that epoxy resins can withstand radiation dosage up to 9.5×10^8 rads without degradation of physical or electrical properties.

2.3 Discussions:

2.3.1 Radiation:

The radiation withstand capability of the complete electrical penetration is based on the following materials:

a. Epoxy - As discussed in Paragraph 2.2.4, it is confirmed that epoxy can withstand radiation dosage up to 2.5×10^8 R which meets the requirement of 1.373×10^7 R.

b. Textolite - Paragraph 2.2.3(b) confirms that Textolite is similar to epoxy laminate with glass fabric and possesses radiation withstand properties even better than 2.5×10^8 R and meets the requirement of 1.373×10^7 R.

c. Chico "A" - Paragraph 2.2.3(a) confirms that Chico "A" is similar to Portland Cement which being inorganic in nature is not subject to radiation damage and, therefore, meets the requirement of 1.373×10^7 R.

d. Ozite "B" - Ozite "B" is an asphaltic based hydrocarbon manufactured by G&W Electric Specialties Corporation. Reference 5 document, Pages 456 through 459, indicates that such hydrocarbons are not affected at radiation levels below 4.5×10^8 rads.

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2.3.2 Simulated service conditions and test duration:

Based on the following facts, it is concluded that the LOCA test duration of 23-1/2 hours (Document References 1 and 2) covers the entire time until the conditions return to essentially ambient.

a. Analysis of containment parameters indicates dwell at maximum temperature of 235°F for 1 hour whereas during the actual LOCA test maximum temperature of 307°F was maintained for 1 hour.

b. Analysis of containment parameters indicates dwell at maximum pressure of 41.7 psia for 1 hour whereas during the actual LOCA test pressure of 41.7 psia was maintained for all the 23-1/2 hours.

Further, it is added here that even though the test duration requirement is 30 days, the analysis of containment parameters shows that following a LOCA, temperature and pressure conditions return to ambient in about 3 days.

2.3.3 Aging:

Based on the very high thermal and radiation withstand capabilities (listed below) of various components used in the fabrication of complete penetration assembly, it is concluded that these penetration assemblies are not susceptible to thermal aging and are considered to be qualified for 40-year life requirement at Big Rock Point Plant.

a. Epoxy and Textolite (Laminated epoxy) - Suitable for continuous operation at service temperatures of 400°F and can withstand radiation dosage of 2.5 x 10^8 rads.

b. Chico "A" - Being similar to Portland Cement is not susceptible to radiation or thermal aging.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous Electrical	Operating Time	30 Days	30 Days	Section II.D	1, 4	Evaluation	
Plant I.D. Number: Penetrations Type 10, 11, 12	Temperature (°F)	235	340	Section II.E	1, 2, 4	Test and Analysis	
Component: Electrical Penetrations; Manufacturer: Conax Corp	Pressure (PSIA)	41.7	74.7	Section II.E	1, 2, 4	Test and Analysis	
Model Number: 7626-10000-01,02,03	Relative Humidity (%)	100	100	Section II.E	1, 2, 4	Test and Analysis	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Transmit Electrical Signals/Control & Instrumentation	Radiation (Rad)	1.373×10^7	2×10^8	Section II.E	1, 4	Test and Analysis	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	3, 4	Test and Analysis	
Location: Containment; Elevation: >590	Submergence	Not Subject To Submergence	-	-	-	-	

Flood Level Elevation 590
Above Flood Level: Yes: X No:

DOCUMENTATION REFERENCES	NOTES
1. Conax Corporation Report #IPS-389 for Qualification of Conax Penetrations for Dresden Nuclear Power Station.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Letter from W S Rautio, Conax Corporation to C A St Onge, Bechtel Power Corp, 10/30/80. 3. Conax Corporation Report #IPS-325 for Design Qualification Material Test Report for Materials Used in Conax Electric Penetration Assemblies. 4. Conax Corporation Report #IPS-107 for Electrical Terminations Subjected to Design Basis Accident Environment for North Anna Power Stations I and II.	

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EQUIPMENT DESCRIPTION
System: Miscellaneous Electrical
Plant I.D. Number: Penetration Type 8
Component: Electrical Penetration
Manufacturer: Amphenol - Borg
Model Number:
Purchase Order Number: 3159 - E-21
Function/Service: Transmit Electrical Signals/Instrumentation (Coaxial)
Accuracy: Spec: Demo:
Location: Containment
Elevation: >590
Flood Level Elevation 590 Above Flood Level: Yes: X No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	30 Days	Section II.D	1, 2	Analysis	
Temperature (°F)	235	290	Section II.E	2	Analysis	
Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Analysis	
Relative Humidity (%)	100	100	Section II.E	Sheet 2	Analysis	
Spray	Not Subject To Spray	-	-	-	-	
Radiation (Rad)	1.373×10^{7}	2.5×10^{8}	Section II.E	1	Analysis	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1, 2 & Section IV	Analysis	
Submergence	Not Subject to Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. "Effects of Radiation on Materials and Components" by Kircher & Bowman.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. "Manual of Process Engineering Calculations" by Clarke & Davidson. 3. CRC Handbook of Chemistry and Physics 53rd Edition 1972-73.	

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Component Electrical Penetrations (Types 10 Thru 12)

Type 10: Electrical Penetration with one (1) feedthrough module for 19/C #10 AWG conductors used in control circuits and one (1) feedthrough module for 20/C #16 AWG D-TP used in instrumentation circuits.

Type 11: Electrical Penetration with two (2) feedthrough modules used in 19/C #10 AWG conductors for control circuits and one (1) feedthrough module for 20/C #16 AWG D-TP used in instrumentation circuits.

Type 12: Electrical Penetration with two (2) feedthrough modules for 19/C #10 AWG conductors used in control circuits; two (2) feedthrough modules for 20/C #16 AWG D-TP used in instrumentation circuits and one (1) 13/C #12 AWG used in instrumentation circuits.

2.0 The subject electrical penetrations are used for Class 1E as well as Nonclass 1E circuits inside the containment. This qualification report, therefore, includes both the functional and the pressure boundary integrity of these penetrations.

2.1 Electrical Penetration Assemblies

The subject Conax electrical penetrations are comprised of a seal body, a conductor feedthrough subassembly, an MK compression ferrule, MK compression cap, terminal blocks, enclosures at each and heat shrink polyolefin tubing.

The seal bodies are fabricated from stainless steel and provide a seal housing for the feedthrough assembly. The feedthrough assemblies are composed of insulated solid conductors, resilient thermoplastic sealants.

The complete assembly is designed to interface with the penetration nozzles which are part of the containment vessel barrier. The nozzle is a Schedule 40, 8" (nominal) OD seamless carbon steel pipe, 12" in length, welded to reactor building shell. The penetration assemblies are attached to the containment vessel nozzle by welding. Each penetration assembly is equipped with two terminal boxes, one on each end.

2.2 The qualification information of the subject electrical penetrations is based on the following documents and discussions.

2.2.1 Document Reference 1 includes Qualification Report IPS-389 for Conax penetrations used in Dresden Nuclear Plant which are similar to the Conax penetrations used at Big Rock Point Plant. The similarity between the electrical penetrations, Types 10 thru 12, and the penetration assembly specimens covered by above report is based on the following facts:

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a. Both of these types of penetrations are designed by Conax Corporation.

b. Similar configurations of feedthrough modules (20/C #16 AWG, 19/C #10 AWG and 13/C #12 AWG) have been used in both of these penetrations.

c. Conductors of feedthrough modules used in both types of penetrations are insulated with similar insulating materials such as Kynar & Kapton.

d. Conax Corporation, in Document Reference 2, has certified that production procedures used for the assembly of both types of penetrations are exactly the same.

e. Conax Corporation, in Document Reference 2, has also certified that similar types of insulating materials have been used for the assembly of both types of penetrations.

During environmental qualification testing covered by Report IPS-389, the Conax penetration assembly, except for terminal block enclosures at both the ends, were first irradiated to 2 x 10^8 rads and then subjected to steam exposure test (to envelope temperature profile given in Figure A1 of IEEE-323, 1974) at 340°F, 60 psig (74.7 psia) and 100% relative humidity. At the end of this LOCA test which continued for 30 days, the electrical penetration assemblies successfully withstood the dielectric and insulation resistance tests.

2.2.2 Document Reference 2 is a letter from W S Rautio of Conax Corporation which confirms the similarity between the Conax electrical penetrations used at Dresden Nuclear Plant to the Conax electrical penetrations used at Big Rock Point Plant.

2.2.3 Document Reference 3 is a Conax Corporation Report IPS-325 which concludes that various insulating materials like polysulfone, kapton, polyolefin with nuclear grade adhesive (WCSF-N), etc, are qualified for more than 40-years' life.

2.2.4 Document Reference 4 is a Qualification Report IPS-107 for similar electrical terminations subjected to Design Basis Accident Environment for North Anna Power Stations I & II. The similarity between the electrical terminations (for penetration Types 10 thru 12) to the specimen terminations for North Anna Conax Corporation penetrations is based on the following facts.

a. Both of these types of terminations are designed by Conax Corporation as a part of complete penetration assembly.

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b. Conax Corportion in Document Reference 2 has certified that production procedures used for the assembly of terminations in both cases are the same.

c. Conax Corporation in Document Reference 2 has also certified that insulating materials used for the assembly terminations in both cases are the same.

During environmental qualification testing covered by Report IPS-107, the terminations (which include the terminal block enclosures used at both ends of the Conax penetration) were irradiated to 2.5×10^7 rads and then subjected to steam exposure test at 290°F, 57.5 psig (72.2 psia) and 100% relative humidity. At the end of this LOCA test which continued for 10 days, the terminations successfully withstood the dielectric and insulation resistance tests. Analysis of containment parameters shows that following a LOCA, temperature and pressure conditions return to ambient in about 3 days. Therefore, it is concluded that LOCA test duration of 10 days covers the entire time until the conditions in the containment return to essentially ambient.

The Report IPS-107 also indicates that one enclosure containing the terminations was subjected to thermal aging at 300°F for 74 hours which conservatively qualifies it to 22 years' life. Because of the fact that these Conax penetrations were installed after 14 years of the start of commercial operation of the Big Rock Point Plant, it is concluded that the enclosures at both ends of the penetrations qualify for the entire 40-years' life of the Plant.

Sheet 2

Component Electrical Penetration (Type 8)

Type 8: Coaxial Cable Penetration Assembly

2.0 The subject electrical penetrations are used for non-Class 1E circuits inside the containment. This qualification report, therefore, is submitted to prove the pressure boundary integrity of the penetration.

2.1 Coaxial Cable Penetration Assembly

This consists of an 8" pipe with a flange on the end to which a plate is bolted. The plate holds the coaxial cable connectors. The complete unit, as such, is welded to 8" IPS Schedule 40 nozzle, 12" long which, in turn, is welded to reactor building shell.

The connectors on the plate have 304 stainless steel shell which is press fit into and then welded to the plate.

These connectors have inserts made of glass material and are fused in place by high heat process to achieve a pressurized seal. These inserts are suitable to handle three RG59 B/U cables. In addition, these connectors are provided with solder type mating connector with cable clamp and a protective cap on either side.

Therefore, the pressure boundary integrity of the complete penetration is based on the qualification of following components which are further discussed in subsequent paragraphs:

Glass Preform

The glass preform which forms the primary pressure boundary for the penetration is made from Corning #9010 glass and is fused in place by high heat process to achieve a pressurized seal. This glass begins to soften at 1160°F and is made up of 67% SiO_2, 7% A_2O, 7% K_2O, 5% Al_2O_3 and 12% BaO_2 plus 2% other. Document Reference 1 shows that this glass is similar to both silica glass and soda-lime glass. The table on Page 370 of Reference 1 indicates that silica glass is good to 2.5×10^8 rads. Further, based on the method of placement and the thickness of the glass, it is concluded that it can withstand 27 psig (41.7 psia).

Shell to Plate Assembly

The connector shell is made of 304 stainless steel which is first press fit into the place and then welded. Such metal-to-metal assemblies should be able to withstand LOCA conditions of 235°F and 41.7 psia for a period of 30 days.

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Gasket

Upon analysis it has been determined that the gasket used between the mating surface of the fixed and the removable connectors is made of either the silicon rubber or the natural, butyl styrene butadiene rubber. Document Reference 1 on Pages 118-129 indicates that all rubbers can withstand at least 2×10^7 rads. Figure D-19 on Page 73 of Document Reference 2 shows that butadiene rubber (BUNA S) can withstand continuous temperature of 290°F.

2.2 The qualification information of the subject coaxial cable electrical penetration is based on the following documents and discussions (in Paragraph 2.1).

2.2.1 Document Reference 1 on Pages 118 thru 129 and Page 370 indicates the radiation withstand capabilities of various rubbers and glass.

2.2.2 Document Reference 2 on Page 73 (Figure D-19) confirms that butadiene rubber (BUNA S) can withstand continuous temperature of 290°F.

2.2.3 Document Reference 3 on Page F-134 indicates that the melting point for glass used for sealing applications is not less than 1300°F.

2.3 AGING

Document References 1 and 2 provide evidence that silica glass or soda-lime glass have much higher threshhold for nuclear radiation (up to 2.5×10^8 rads) exposure than other dielectric materials and have very high temperature withstand capabilities (continuous operation of temperature up to 1000°F). Therefore, it is concluded that this fused glass is not susceptible to degradation due to thermal or radiation aging and is considered to be qualified for 40-years' life at Big Rock Point Plant.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Electrical	Operating Time						
Plant I.D. Number: Electrical Type 9 Penetration	Temperature (°F)						
Component: Electrical Penetration	Pressure (PSIA)						
Manufacturer:							
Model Number: Type 9	Relative Humidity (%)						
Purchase Order Number:	Spray						
Function/Service: R & D Penetration For Instrumented Fuel Assembly Wires	Radiation (Rad)						
Accuracy: Spec: Demo:	Aging						
Location: Containment Elevation: >590	Submergence						
Flood Level Elevation 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES

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Component Electrical Penetration Type 9

This penetration, Type 9, was used in a research and development program to instrument fuel assemblies. The penetration provided the electrical path from the fuel assembly to the computer.

This penetration was apparently somewhat different from other types. At the present time, the records for this penetration are being sought so that a detailed analysis can be made for qualification. The qualification is directed toward containment boundary rather than electrical operability.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: FT-2161 through 2164; Component: Flow Transmitter	Temperature (°F)	235	288	Section II.E	2, 3, 4, 5	Test and Evaluation	
Manufacturer: ITT Barton	Pressure (PSIA)	41.7	74.7	Section II.E	2, 3, 5	Test and Evaluation	
Model Number: 386	Relative Humidity (%)	100	100	Section II.E	2, 3, 5	Test and Evaluation	
Purchase Order Number:	Spray	Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Function/Service: Flow Indication Containment and Core Spray	Radiation (Rad)	7.3×10^5	2.16×10^8	Section II.E	1, 3	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment; Elevation: 625 Except One at 578	Submergence	Yes	Sheet 2	Section II.E	Sheet 2	Evaluation	
Flood Level Elevation: 590; Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. Letter J Bruno, Westinghouse Process Control Systems, to E A Lommatsch, ITT Barton, January 21, 1971.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. F-C2667 Performance Test of Three Differential Pressure Transmitters in a Simulated Reactor Containment Post-Accident Steam Environment, Franklin Institute Research Laboratories, November 1969. 3. Letter R M Marusich to J Doyon, ITT Barton. 4. ITT Barton Product Bulletin 332-2. 5. Letter from J Doyon, ITT Barton, to R Marusich, October 27, 1980, concerning similarity between Model 332 and Model 386.	

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Component Flow Transmitter FT-2161 Through FT-2164

The flow transmitters are Model 386. The flow transmitter is used to provide indication of core spray flow and also to detect if a core spray line is broken.

The environmental test used a Model 332. Model 332 differs from Model 386 only in housing type. However, for the environmental test, the Model 386 housing was used on the Model 332 and, therefore, the test specimen is identical to that installed. The test sequence was to use steam to pressurize a test chamber to 60 psig. This pressure and a temperature of 288°F was maintained over a two-hour test period and then cooled. The test instruments were read twice during the test and the readings matched that of the transmitter outside the chamber. The instruments were also separately tested for their ability to withstand radiation. They were exposed to 10^6 rads/h for 216 hours and passed. The total integrated dose far exceeds the qualification requirements. The transmitters are housed in a metal enclosure so that sprays will not affect them. Two hours after a LOCA, the temperature is at or below 160°F. The manufacturer states that the operational range of this transmitter is -40°F to 160°F. Therefore, the transmitter is able to withstand the effects of the LOCA because it was tested in a more severe environment and for the time period over which the containment conditions are more severe then the operational limits. No information has been found concerning age sensitivity. Should these transmitters fail, due to age degradation, failure will, in all probability, occur after the operator has ascertained the integrity of the core spray line. Once that is accomplished, core spray flow indication is not required for plant shutdown. However, since core spray flow indication may be useful to the operator, age sensitivity will be investigated and appropriate action (partial replacement, full replacement, etc) will be taken before June 30, 1982, assuming no procurement problems.

One of the flow transmitters will become submerged during the accident. The flow transmitter is considered qualified for submergence for the following reasons. Employees at the plant have stated that the transmitter and its leads are sealed. The transmitter was tested in a steam environment of 60 psig. This is far greater than the pressure the instrument will see even if submerged. It is believed that if steam at 60 psig cannot enter the transmitter to the extent of causing failure, then water at 27 psig will not result in failure.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	21 Hours	See Sheet 2	Section II.D	Section II.D	Evaluation	
Plant I.D. Number: LS-3562 LS-3563 LS-3564 LS-3565	Temperature (°F)	235	See Sheet 2	Section II.E	1, 3, 4	Evaluation	
Component: Level Switch	Pressure (PSIA)	41.7	44.7	Section II.E	1, 3, 4	Evaluation	
Manufacturer: Jo-Bell; Model Number: Type R	Relative Humidity (%)	100	See Sheet 2	Section II.E	1, 3, 4	Evaluation	
Purchase Order Number: M-82	Spray	Lake Michigan Water	See Sheet 2	Section II.E	1, 3, 4, 6	Evaluation	
Function/Service: Containment Water Level Indication	Radiation (Rad)	1.1×10^6	5×10^6	Section II.E	1, 2, 4	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Section IV	Section II.E	1, 3, 4, 5	Evaluation	
Location: Containment; Elevation: 574,579,587,595	Submergence	Yes	See Sheet 2	Section II.F	1, 3, 4, 6	Evaluation	

Flood Level Elevation: 590
Above Flood Level: Yes: No: X

DOCUMENTATION REFERENCES	NOTES
1. Letter R M Marusich, CP Co, to W Maurer, of Jo-Bell Dated October 16, 1980	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Reference Book by Kircher and Brown, 1964, "Effects of Radiation on Materials and Components" 3. Big Rock Point Task Force File K.5.1 4. Big Rock Point Task Force File 182 5. Nuclear Environmental Qualificaton for Palisades PS-1801, et al, Wyle Laboratories, August 4, 1980 6. R M Marusich to R E Schrader, BRP Maru 70-80 Dated October 16, 1980	

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Component Level Switches LS-3562 Through LS-3565

Level Switches LS-3562 through LS-3565 are used to provide containment water level indication. These four switches are located at elevations 574', 579', 587' and 595'. The switches close a contact when the water level reaches them. The electrical scheme is such that the switches are wired in parallel to the power supply. Failure of one switch will not impair the rest of the circuit.

The internal switch and trip mechanism, rated to 160°F, are housed in an explosionproof steel housing. The containment water temperature is above 160°F for, at most, 5.5 hours. The atmosphere temperature is above 160°F for a longer period of time. The housing will cause the temperature of the internal switch to lag that of containment such that the switch temperature is above 160°F for a shorter time. The switch is all metal construction except for an asbestos gasket and neoprene wires. The switch is rated at 44.7 psia, which is in excess of the maximum containment pressure. The switch is sealed with the asbestos gasket while the electrical connections are sealed with Chico sealing compound placed in a sealing cage. It should be noted that these switches were designed to be submerged. Therefore, it is not expected that sprays, humidity or submergence will affect these switches. The known organic materials, asbestos and neoprene are acceptable for service to a radiation dose of at least 5×10^6 rads, whereas the 24-hour gamma dose in water is 1.1×10^6 rads. Further analyses of these materials may be required for determination of their sensitivity to thermal aging as was discussed in Section IV of this report.

In summary, no type test of this equipment is available as required by the DOR guidelines.

The reader should note that these switches may be considered as backup indication to LT-3171. It is currently planned to install a new level indicating system in the fall of 1981.

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EQUIPMENT DESCRIPTION

System: Post-Incident

Plant I.D. Number: LS-RE06A LS-RE20A LS-RE06B LS-RE20B

Component: Level Switch

Manufacturer: Yarway

Model Number: 4320 PE

Purchase Order Number:

Function/Service: Reactor Scram on Low Steam Drum Level

Accuracy: Spec: Demo:

Location: Containment

Elevation: 621

Flood Level Elevation: 590

Above Flood Level: Yes: X No:

	ENVIRONMENT		DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	10 Minutes	10 Minutes	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	217	269	Section II.E	2 and Sheet 2	Test and Evaluation	
Pressure (PSIA)	~16.5	44.7	Section II.E	2 and Sheet 2	Test and Evaluation	
Relative Humidity (%)	100	100	Section II.E	2 and Sheet 2	Test and Evaluation	
Spray	Lake Michigan Water	Lake Michigan Water	Section II.E	Sheet 2	Evaluation	
Radiation (Rad)	4.9×10^3	4.9×10^3	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Submerged	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Letter, Consumers Power Company to NRC, D P Hoffman to J G Keppler 8/26/80. 2. NUS test on Yarway Model 4416C NUS-TM-ED-116 March 1975.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Yarway test on Model 4418TC March 1979. 4. Yarway test on Model 4418EC March 1973.	

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Component Level Switches LS-RE06A,B and LS-RE20A,B

Level switches LS-RE06A,B and LS-RE20A,B provide the necessary signal to the reactor protection system for a reactor scram due to low water level in the steam drum. In addition to the function described above, the LS-RE20A,B units are fitted with a differential transformer and armature rod assembly driven by the primary instrument pointer to provide that an electrical signal can be sent to slave indicators located in the control room.

The required operating time for these instruments has been established at 10 minutes. The basis for this time is that there are some small sized steam line breaks that may occur which will not provide an automatic scram due to low steam drum level or containment high pressure. This is because (1) the water inventory lost through the break is maintained by the feedwater system and (2) the containment is normally vented to the atmosphere via the stack such that for small breaks the containment high pressure set point may not be reached. These break sizes involve leakages of 75#/s or less. No credit for operator action is taken for the first 10 minutes of a steam line break event. After 10 minutes it is reasonable to expect the operator to take the action necessary to manually shut down the reactor. For break sizes resulting in flow rates greater than 75#/s, the level switches must remain operable to automatically scram the reactor. A discussion of the small break scenario was provided to NRC on 8/26/80 as noted in Reference 1. It is worthwhile to note that from the data submitted as Reference 2, the most limiting steam line break size results in a flow rate of 75#/s. With this flow rate, containment temperature was calculated to reach 217°F after 10 minutes rather than the peak LOCA temperature of 235°F.

If the reactor scrams from high containment pressure via PS-664 through PS-667, then there is no need for LS-RE06A,B and LS-RE20A,B since they are only needed for the scram function. Thus, the pressure that the level switches need be qualified for is that corresponding to the set point of the pressure switches PS-664 through PS-667, namely 16.4 psia.

The maximum radiation dose the level switches must reasonably endure is 4.9×10^3 rads. The basis for this dose value is that if the full 10-minute time interval is needed, the drum level is being maintained by normal feedwater addition. In this case, there would be no core uncovery with resulting fuel damage. The radiation dose assigned for a steam line break in the pipe tunnel assumed a total loss of the primary coolant inventory with the highest level of coolant activity allowed by Technical Specifications. The small steam line break event is similar to the MSLB outside containment with respect to the source of radiation and therefore the pipe tunnel dose was used. For the larger break sizes, the required operating times are very much shorter, thus less dosage is incurred by the level switches.

The level switch vendor, Yarway, was asked to provide a listing of organic materials used in the Model 4320PE level switches. The vendor responded with the following for the 4300 series:

Sheet 2 (Contd)

1. Diaphragm - Neoprene Coated Dacron Fabric

2. Backing Plate Gasket - Klingerit (Compressed Asbestos)

3. Dust Plug Grommet - Neoprene

4. Light Diffuser - Molded Translucent Plexiglas

5. Front Cover - Molded Clear Plexiglas

6. Dial - Translucent White Vinyl

NOTE: Items 4, 5 and 6 above have been replaced with Lexan at Big Rock Point.

7. Casing Gasket - Cork and Rubber Compound DK-149

8. Switch Spacers - Laminated Melamine

9. Switch - Mercury

10. Relays - BO Type - Allied Control Co

Other organic materials found in the armature rod assembly and differential transformer and guide assembly include:

11. Armature Bushing - Synthane Grade L

12. Differential Transformer and Guide Assembly - Copper Wire With Silver Plated Teflon Insulation (MIL-W-16878, Type E), Epoxy CPI-4270 Hysol, Markol Corp Shrink Tubing HT105C and HF105

13. Terminal Blocks - Molded Phenolic Base

14. Coil Bobbin - Nylon

15. Winding - Nyclad Wire

16. Disc - Synthane Grade LE

17. Sealer - Epoxy

18. Insulator - Fish Paper 0.010" Thick

The level instruments need to be qualified for a rise in temperature from ambient to 217°F over a period of 10 minutes. The Model 4300 series are similar in design and materials of construction as the Model 4400 series. Various tests (References 2, 3 and 4) were performed on different 4400 series instruments covering a range of peak temperatures from 212°F to 269°F, with a peak pressure of 44.7 psia and 100% relative humidity. These tests provide

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the necessary assurance that LS-RE06A,B and LS-RE20A,B will remain operable until their intended function is complete.

The spray water in containment is taken from Lake Michigan which is relatively pure. The level instruments are located on a concrete wall below the containment spray header; however, the nozzles are oriented in the horizontal direction so that the instruments will not be subjected to direct impingement of the spray. Further, the instruments are mounted in a metal cabinet with a top that will protect the instrument from spray.

The most susceptible material to the radiation dose the instruments must endure is the silver plated Teflon wire. According to the Nuclear Engineering Handbook by H. Etherington, et al, Page 10-144, Teflon-insulated wire suffers a 25% change in elongation at a radiation dose of 3 x 10^4 rads. Thus, these level switches are deemed qualified for radiation.

Although the ambient temperatures are on the average low (70°F), it is recognized that some of the components in the instruments may be susceptible to thermal aging. The LOCA event will contribute little to the overall aging because of the short time the equipment is required to function. Further investigation will be performed to determine which materials are age sensitive and whether or not failure of these parts will affect instrument operation.

To summarize, it is concluded LS-RE06A,B and LS-RE20A,B will satisfactorily perform their intended function during the LOCA event.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident/ Reactor Protection	Operating Time	1 Hour	1 Hour	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: LS-RE09 A Through H	Temperature (°F)	235	269	Section II.E	1, 5 and Sheet 2	Test	
Component: Reactor Water Level	Pressure (PSIA)	41.7	44.7	Section II.E	1, 5 and Sheet 2	Test	
Manufacturer: Yarway Corp; Model Number: 4420C	Relative Humidity (%)	100	100	Section II.E	1, 5 and Sheet 2	Test	
Purchase Order Number:	Spray	None	None	Section II.E	-	-	
Function/Service: 1. Core Spray Valve Open Permissive 2. Reactor Scram and Cont Isolation	Radiation (Rad)	$<2.0 \times 10^5$	2×10^5	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo: Location: Containment	Aging	40 Years + 1 Hour of LOCA	40 Years + 1 Hour of LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Elevation: 590.5	Submergence	No	-	-	-	-	
Flood Level Elevation: 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Test Report NUS-TM-ED-116 March 1975 for Yarway Model 4416C. 2. Letter R B Cherba to C J Hartman 5/20/75.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Letter R B Cherba/H J Palmer to C J Hartman 5/19/75. 4. Deviation Report DAD-100-678-2, 10/5/78. 5. Engineering Research Laboratory Project 378018.10A.	

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Sheet 2

Component Level Switches LS-RE09A Through H

The LS-RE09A through D level switches are used for three purposes: (1) to provide a reactor scram on low reactor water level, (2) to provide a containment isolation signal on low reactor water level and (3) to provide an open signal to the primary core spray valves MO-7051 and MO-7061. LS-RE09E through H are used to provide an open signal to back up core spray valves MO-7070 and MO-7071 on low reactor water level. The time required for these switches was established at one hour. The basis for this time is that it is the longest time before the low reactor water level set point is reached due to a small break (0.008 ft^2) LOCA. In addition, the level switches must operate in conjunction with PS-IG11A through H to open the core spray valves. The PS-IG11A through H instruments close contacts in the core spray valve electrical scheme when reactor pressure reaches ≤ 200 psig which are in series with the LS-RE09A through H contacts. For the 0.008 ft^2 break LOCA, the reactor depressurizing system (RDS) must first operate to blow the primary system down in order to reach ≤ 200 psig. In this scenario, the RDS will actuate when the low reactor water level set point (the same set point as for LS-RE09) is reached via LT-3180 through LT-3183. IT SHOULD BE NOTED, however, that the reactor operator is directed by procedure to manually initiate the RDS if he deems it necessary. This being the case, it is almost a certainty that the operator will actuate RDS at some time before one hour has elapsed, thus fulfilling the requirements to open the core spray valves. The point being made here is that in all probability the LS-RE09 will have to function in a shorter time and endure a less severe environment than that shown on the qualification sheet. For larger break size LOCA, the time required for LS-RE09 to function automatically is reduced less than two minutes.

The Yarway LS-RE09A through H level switches were modified in 1975 subsequent to a test performed on a similar Model 4416C by the NUS Corporation for Consumers Power (Reference 1). These modifications included (1) removal of the plastic dial cover, dial and diffuser plate, (2) replacement of the mercury switches with ACRO switches, (3) removal of the light power socket, (4) replacement of plastic plugs on the top of the unit with gasketed metal plugs and (5) installation of a gasketed aluminum cover (References 2 and 3). The reasons for these modifications are as follows:

1. After exposing the test instrument for approximately three and one-half hours to a saturated steam environment with the temperature and pressure at 275°F and 29 psig, the low-level switch failed to trip because the instrument pointer became bound on the dial/light diffuser that had weakened and sagged due to the prolonged high temperature.

2. The switches were replaced with ACRO switches due to seismic considerations.

3. The light power socket was removed in order to avoid subjecting the instrument case to a pressure which could cause it to collapse.

Sheet 2 (Contd)

4. The plastic plugs were changed to metal gasketed plugs to temperature considerations.

5. The gasketed aluminum cover was installed for protection of the open face of the instrument.

On October 5, 1978, it was discovered that the Yarway model 4416C was tested with mercury switches instead of the ACRO switches (Reference 4). Corrective action taken by Consumers Power was to send an ACRO switch in stock that was procured for the LS-RE09 to the Consumers Power Company System Protection and Laboratory Services Department for a four-hour test with the temperature and pressure at 132°C (269°F) and 29 psig with 100% relative humidity. The switch was operated hourly. The switch successfully passed this test (Reference 5).

Although it is true that the level switch tested by NUS is not identical to the level switches installed at Big Rock Point, the manufacturer has stated that the only differences between the Model 4416C and the 4420C were in the front and rear housing materials (ie, naval brass versus bronze) and the switches (ie, mercury versus ACRO). It is concluded, therefore, that the test reports listed as References 1 and 5 are acceptable documentation to qualify the LS-RE09A through H, Yarway Model 4420C, as modified for the temperature, pressure and humidity conditions that they will be subjected to during their required function time in the event of a LOCA.

The containment spray water at Big Rock Point is taken from Lake Michigan. It contains no chemicals and is relatively pure water. The LS-RE09 are located in the fuel pit heat exchanger room and, thus, will not be subjected directly to the spray.

The instrument supplier was contacted to provide a list of organic materials used in the 4400 series. The list provided includes:

1. A diaphragm made of Dacron fabric and EPT elastomer.

2. An O-ring on the range adjustment screw made of BUNA N.

3. A backing plate seal made of ethylene-propylene terpolymer.

4. A dust plug made of neoprene.

5. A light diffuser, cover and dial made of Plexiglas and white vinyl (these are no longer installed in LS-RE09A through H).

6. Casing gasket made of cork and rubber compound DK-149.

7. Switch spacers made of laminated melamine.

8. ACRO switch made of molded diallyl phthalate.

9. Insulating tubing made of No 10 clear-MIL-I-631C.

Sheet 2 (Contd)

10. Terminal blocks made of molded phenolic base.

11. A disc at the bottom of the coil bobbin made of synthane Grade LE.

12. The coil bobbin made of nylon.

13. Sealer made of epoxy.

14. Insulator 0.010" fish paper.

The radiation dose at the containment centerline after two hours is 2×10^5 rads. As mentioned earlier, these instruments are unlikely to be required for even one hour. In addition, the instruments are mounted on a substantial concrete wall that would allow one to reduce the dose by a factor of 2. Lastly, although the radiation dose listed assumes 100% core melt, in reality these instruments will perform their functions prior to core uncovery; thus, the actual core damage will probably be no more than that allowed by Technical Specifications for normal operation. According to the Nuclear Engineering Handbook by H Etherington, et al, of the materials commonly used in electrical equipment, motors, valves, relays, switches, transformers, etc, the only materials that cannot successfully withstand a dose of 2×10^5 rads are Teflon and thiskol (polysulfide rubber). As these materials are not in those listed for these instruments, they are considered to be qualified for radiation up to and including the time required for operation.

The instruments have not been tested for thermal aging. Referring back to the list of organic materials, the only material expected to be age sensitive is the BUNA N O-ring on the range adjustment screw. From the drawing, this O-ring does not appear to serve an important function.

In summary, it is concluded from the above discussion that the LS-RE09A through H reactor water level switches will adequately perform their intended function during the LOCA event.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: LT-3171; Component: Level Transmitter	Temperature (°F)	235	288	Section II.E	5, 2, 3, 4	Test and Evaluation	
Manufacturer: ITT Barton	Pressure (PSIA)	41.7	74.7	Section II.E	2, 3, 5	Test and Evaluation	
Model Number: 386	Relative Humidity (%)	100%	100%	Section II.E	2, 3, 5	Test and Evaluation	
Purchase Order Number:	Spray	Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Function/Service: Containment Water Level Indication	Radiation (Rad)	7.3×10^5	2.16×10^8	Section II.E	1, 3	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment; Elevation: 595'	Submergence	No	-	-	-	-	
Flood Level Elevation 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Letter J Bruno, Westinghouse Process Control Systems, to E A Lommatsch, ITT Barton, January 21, 1971.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. F-C2667 Performance Test of Three Differential Pressure Transmitters in a Simulated Reactor Containment Post-Accident Steam Environment, Franklin Institute Research Laboratories, November 1969. 3. Letter R M Marusich to J Doyon, ITT Barton. 4. ITT Barton Product Bulletin 332-2. 5. Letter from J Doyon, ITT Barton, to R M Marusich, October 27, 1980, concerning similarity of Model 332 and 386.	

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Component Level Transmitter LT-3171

The level transmitter is a Model 386. The transmitter is used to provide indication of containment water level to the operatore prior to entering the recirculation mode. This transmitter works in parallel with LS-3562 through LS-3565. Model 332 was environmentally tested. The environmental test used a Model 332. Model 332 differs from Model 386 only in housing type. However, for the environmental test, the Model 386 housing was used on the tested Model 332 and, therefore, the test specimen is identical to that installed. The test sequence was to use steam to pressurize a test chamber to 60 psig. This pressure and a temperature of 288°F was maintained over a two-hour test period and then cooled. The test instruments were read twice during the test and the readings matched that of the transmitter outside the chamber. The instruments were also separately tested for their ability to withstand radiation. They were exposed to 10^6 rads/h for 216 hours and passed. The total integrated dose far exceeds the qualification requirements. The transmitters are housed in a metal enclosure so that sprays will not affect them. Five and one-half hours after a LOCA, the temperature is at or below 160°F. The manufacturer states that the operational range of this transmitter is -40°F to 160°F. Therefore, the transmitter is able to withstand the effects of the LOCA because it was tested in a more severe environment and for the time period over which the containment conditions are more severe then the operational limits. No information has been found concerning age sensitivity. Should these transmitters fail due to age degradation, containment level indication can be obtained by use of LS-3562 through LS-3565. As part of the modifications, as a result of TMI, a new fully qualified level transmitter will be installed.

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EQUIPMENT DESCRIPTION

System: Reactor Depressurizing

Plant I.D. Number: LT-3180, LT-3181, LT 3182, LT-3183

Component: Level Transmitter

Manufacturer: Westinghouse

Model Number: 59DP4C997050

Purchase Order Number:

Function/Service: Reactor Water Level Transmitter

Accuracy: Spec:
Demo:

Location: Containment

Elevation: 590.5

Flood Level Elevation: 590

Above Flood Level: Yes: X No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Test and Evaluation	
Temperature (°F)	235	288	Section II.E	1	Test	
Pressure (PSIA)	41.7	73.7	Section II.E	1	Test	
Relative Humidity (%)	100	100	Section II.E	1	Test	
Spray	Lake Michigan Water	1.5% by Wt Boric Acid; NaOH to pH of 10.4	Section II.E	1	Test	
Radiation (Rad)	7.3×10^5	2×10^8	Section II.E	1	Test	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	No	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Qualification testing of Veri Trak Pressure Transmitters for service in a Nuclear Reactor Containment Facility, Franklin Institute Research Laboratories, October 1973, F-C3715.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Westinghouse Information Bulletin IB-101-191, Page 4, June 1974.	

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	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Depressurizing	Operating Time	1 Hour	24 Hours	Section II.D	Sheet 2	Test	
Plant I.D. Number: LT-3184, LT-3185, LT-3186, LT-3187	Temperature (°F)	235	288	Section II.E	1	Test	
Component: Level Transmitter	Pressure (PSIA)	41.7	73.7	Section II.E	1	Test	
Manufacturer: Westinghouse	Relative Humidity (%)	100	100	Section II.E	1	Test	
Model Number: 59DP4C997050	Spray	Lake Michigan Water	1.5 W/O Boric Acid NaOH to pH = 10.4	Section II.E	1	Test	
Purchase Order Number:	Radiation (Rad)	7.3×10^5	2×10^8	Section II.E	1	Test	
Function/Service: Steam Drum Level Transmitter	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Submergence	Not Subject To Submergence	-	-	-	-	
Location: Containment							
Elevation: 621							
Flood Level Elevation 590							
Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Qualification testing of Veri Trak Pressure Transmitters for service in a Nuclear Reactor Containment Facility, Franklin Institute Research Laboratories, October 1973, F-C3715.	
2. Westinghouse IB-101-191, Page 4, June 1974.	

Sheet 2

Component LT-3180-3187

The level transmitters are Westinghouse level transmitters which were tested by Franklin Institute Research Laboratories. The test specimen was the same as that installed with the following exceptions: (1) the test specimen had a -400 to 400 inch WC Type 316 stainless steel diaphragm where the installed transmitter had a 0-60 inch WC Type 316 stainless steel diaphragm, (2) the test specimen had a DC 550 silicone oil fill (1,500 psig) vs DC 550 silicone oil fill (3,300 psig) for the installed transmitter, and (3) the test specimen had no "Cleaning for Nuclear Service" where the installed transmitter did. Since the materials are the same and only their ratings are different, the test specimen and that installed are considered essentially identical.

The test sequence was as follows: Steam was added to the test chamber pressurizing it to 58 psig in 9.3 seconds and 68 psig, 304°F in 49 seconds. These conditions were maintained for 3 minutes and then reduced to 288°F, 59 psig and held for 24 hours. The spray solution was turned on at 56 seconds at a rate of 1 gpm. The solution was 1.5% by weight boric acid buffered with NaOH to a pH of 10. After the test, the samples were irradiated at 1 Mrad/h for 192.6 hours and 0.2 Mrad/h for 38.5 hours for a total of 2×10^8 rad. These test conditions are more harsh than that required for qualification and therefore the guidelines concerning the enveloping of the peaks are met. The test duration is 24 hours. The temperature profile of the accident condition shows that at 24 hours the temperature is below 120°F. The test was shorter than the temperature-pressure envelope required; however, the peak test temperatures and pressures far exceeded the envelope. Thus, the test duration is felt to be adequate to show qualification of these instruments.

Age related information could not be located. The transmitters are in a normal environment of 50 to 90°F (seasonal variation). CP Co will continue to search for information concerning the existence of age sensitive materials or aging qualification. Continued operation is justified by the arguments presented in Section 4.

The test duration is considered to be of sufficient duration and the test conditions exceed those required to demonstrate qualification.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: MO-7051 MO-7061	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Valve Actuator; Manufacturer: Limitorque	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: SMA-00	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	–	–	–	–	
Function/Service: Primary Core Spray Valves	Radiation (Rad)	1.36×10^6	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment; Elevation: 599 Ft	Submergence	Not Subject To Submergence	–	–	–	–	
Flood Level Elevation 596 Ft; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Report F-C4124 performance qualification tests of four motor-operated valves.	

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EQUIPMENT DESCRIPTION

System: Main Steam

Plant I.D. Number: MO-7050

Component: Valve Actuator

Manufacturer: Limitorque

Model Number: SMA-2-60

Purchase Order Number:

Function/Service: Main Steam Isolation Valve

Accuracy: Spec:
Demo:

Location: Containment

Elevation: 612

Flood Level Elevation: 596

Above Flood Level: Yes: X No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject To Spray	-	-	-	-	
Radiation (Rad)	9.1×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Subject To Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Report F-C4124 performance qualification of four motor-operated valves.	

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Component Limitorque Actuators MO-7051, -7061, -7050

These Limitorque valve actuators are all dc operated. The valve motors are Peerless with Class B insulation. The Limitorque actuators are Type SMA. A LOCA qualification test for these valves was conducted by Franklin Laboratories for Consumers Power Company in April 1975. Report F-C4124 (Reference 1) describes the test conditions. Qualification of the three subject valves was based on the testing of a similar unit from the BRP Plant. The unit tested was the same actuator size as MO-7051 and -7061 and all are from the same SMA family. The unit also contained a Peerless Class B insulated dc motor as do both of the core spray motors. MO-7050 has a larger SMA actuator, but contains the same materials as the actuator tested. The FIRL test exposed the operators for 50 minutes to 240°F and 43.7 psia with tap water spray. Because of the location of the valves, containment sprays will not impinge upon them. A near 100% relative humidity (RH) was maintained throughout most, if not all, of the test by the following means: (1) use of saturated steam to obtain the initial temperature and pressure rise to 240°F, (2) use of fine mist sprays over the specimens and (3) use of saturated steam ejections to maintain chamber temperatures.

Another Limitorque SMA actuator was tested in a 36-hour steam/28-hour water spray exposure. This particular actuator was equipped with an ac motor. Both satisfactorily passed the test.

The MSIV (MO-7050) closes on a containment isolation signal, either containment high pressure or low reactor water level, and will not be required to operate after closing. Closing of the valve will be accomplished within ten minutes. The core spray valves (MO-7051 and MO-7061) open within one hour on a low reactor water signal and low reactor pressure interlock. Procedurally, after recirculation has begun (maximum of 21 hours), the core spray valves may be closed if the break has been reasoned to be in this core spray line. Closing the valves would be done in conjunction with keeping the backup (redundant) core spray line in operation to provide core cooling. After closing, it would not be necessary to change the valve lineup.

Radiation and thermal aging qualification testing for the particular Limitorque actuator and Peerless motor has not been done to our knowledge and was not done on the tested actuator. Limitorque Test Report B0009 for a Limitorque SMB-0-25 and Peerless dc Class H insulated motor aged the unit at 180°C for 100 hours and 2,000 operating cycles and subjected the operator to 1×10^7 rads of gamma radiation. This was followed by a 25-hour LOCA simulation. There is, of course, no basis to assume the Test Report B0009 qualifies the installed valves by similarity except for both motors operating by direct current.

By a general idea of the materials composing the actuator and motor, they would be able to withstand radiation doses of at least 4.0×10^6 rads based on Nuclear Engineering Handbook, H Etherington, et al. Depending on the susceptible materials, they would experience degradation of their physical properties with a dose in excess of 4×10^6 rads. The normal environmental

Sheet 2 (Contd)

dose rate is 10 R/h during power operation for the core spray valves. Twenty years at 10 R/h plus a shutdown dose rate of 350 mR/h 60% of the time yields a current dose of ~ 1.4×10^6 rads. With a conservatively chosen damage value of 4×10^6 rads, the valve actuators are determined to operate in the radiation environment they experience in a post-LOCA environment.

Thermal aging of the actuator would be accelerated in the event of a LOCA or MSLB inside containment. Such an effect after a 40-year life has not been simulated by type testing. The valve tested in the 1975 test, Report F-C4124, after thirteen years of actual inservice life with normal temperatures ranging from 50°F to 90°F while shut down and 170°F to 200°F when operating, remained operable during the 50-minute LOCA simulation. The normal temperature the core spray valves and MSIV are in is 50°F to 90°F while in shutdown and 100°F to 140°F while the Plant is in operation.

Based on the testing, Report F-64124, and because the test valve had been thermally aged at an accelerated rate during its service life yet passed the 50-minute LOCA simulation, these valves are considered acceptable for present use. However, no thermal and radiation aging data is available for these valve actuators. Because the aging phenomenon is not a well-known entity and due to the LOCA simulation testing not encompassing the full LOCA envelope, these actuators and motors will either be replaced or rebuilt and requalified by June 30, 1982.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION

System:
Post-Incident

Plant I.D. Number:
MO-7064

Component:
Valve Actuator

Manufacturer:
Rotork

Model Number:
14 A Syncroset

Purchase Order Number:
S/N 53604

Function/Service:
Containment
Spray Valve

Accuracy: Spec:
Demo:

Location:
Containment

Elevation: 625

Flood Level Elevation: 596
Above Flood Level:
Yes: X No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Test and Evaluation	
Temperature (°F)	235	Sheet 2	Section II.E	1	Test and Evaluation	
Pressure (PSIA)	41.7	Sheet 2	Section II.E	1	Test and Evaluation	
Relative Humidity (%)	100	Sheet 2	Section II.E	1	Test and Evaluation	
Spray	Lake Water	Sheet 2	Section II.E	1	Test and Evaluation	
Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Test and Evaluation	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Test and Evaluation	
Submergence	Not Subject To Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Report F-C4124 performance qualification tests of four valve motor operators. 2. Rotork Test Report No TR422, Heat Aging Class B, NA2 motor for "Outside Containment Safety-Related Duty."	

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Owner: Consumers Power Company
Facility: Big Rock Point
Docket: 50-155

EQUIPMENT QUALIFICATION REPORT

Component Sheet No.:
Revision:
Date:

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Letter Rotork RHA to CP Co JLK dated October 17, 1980 with attached summary for "A" range actuators. 4. Rotork Test No N/14/2 "Actuator Radiation Subjection." 5. Limitorque Qualification Test Reports No B0003 and No B0009.	

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Sheet 2

Component Rotork Valve Actuator MO-7064

This Rotork actuator contains a dc motor with Class B insulation and weatherproof enclosure. The actuator was installed in the Plant in 1970. The valve is actuated by containment high pressure and presently has a 15-minute time delay. The valve will be required to actuate intermittently in the post-LOCA environment until the containment atmosphere returns to ambient. From the LOCA pressure-temperature envelope this will be approximately 1 day. Subsequently, the valve will remain closed.

Report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators" subjected an exact type of actuator and motor to a 36-hour LOCA simulation. The simulation followed the LOCA envelope curve for this period of time. The test also included 24 continuous hours of spraying for the first 24 hours and then intermittent 1 hour spraying at hours 27, 31 and 35. Relative humidity was maintained at or near 100% by (1) use of saturated steam to obtain initial temperature and pressure rise to 240°F, (2) use of fine mist water over the test specimen, (3) use of saturated steam ejections to maintain chamber temperatures. Radiation and thermal aging were not included as part of the test procedure. Rotork has conducted testing for aging qualification. Rotork Technical Report TR422 subjected an NA type actuator with a Class B insulated motor to an aging test which simulated 48 years at 60°C. The unit in the test was placed in a furnace for 100 hours at 180°C. 48 years at 60°C was determined by the 10°C rule. TR422 did not include a LOCA simulation, and although an NA type actuator was used in the test the only difference between the NA and 14A units, according to Rotork, is the motor housing material, cast iron versus cast aluminum. The results of the test, Rotork states, and the electrical characteristics will be exactly the same for the 14A as the NA type units. Rotork Test No N/14/2 used a prototype unit built for material evaluation for the Standard "A" and NA1 components. The actuator to MO-7064 being a Standard "A" type unit. The test results showed the "A" range components capable of withstanding 30 megarads during their 40-year life, according to Rotork. The test unit, however, was equipped with a Class H insulated motor. Limitorque actuators (Report B0003) with Class B, ac motors have been irradiated to 20 megarads and been subjected to outside containment HELB conditions for a 16-day period and have operated satisfactorily.

Another limitorque test (Report B0009) subjected a Class H, dc motor to 10 megarads of gamma radiation and followed with a 25-hour HELB, simulation. The unit was also preaged at 180°C for 100 hours and satisfactorily passed the test.

The 30-day radiation dose given on the qualification sheet can be divided in half as the actuator is mounted adjacent to a 3.5-foot thick concrete wall. An integrated dose of 3.7×10^5 rads is within the qualification dose of most materials. A one-day dose of 4.9×10^5 halved giving 2.5×10^5 is what the actuator will see during the time required to operate. This is also well within almost any material radiation resistance.

Sheet 2 (Contd)

In summary, based on the testing of similar type units for radiation and thermal aging and the test conducted by CP Co, the actuator is expected to operate for the required one-day period. After one day, the valve will not be required to operate and will remain closed. The motor starter is installed in the station power room outside containment and will not be subjected to a harsh environment; therefore, a misoperation due to failure of the starter is not a credible failure. The actuator is considered acceptable for use.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION

System: Post-Incident

Plant I.D. Number: MO-7066

Component: Valve Actuator

Manufacturer: Limitorque

Model Number: SMA-00

Purchase Order Number:

Function/Service: Post-Incident HX Cooling Water

Accuracy: Spec: Demo:

Location: Core Spray Room

Elevation: 586

Flood Level Elevation: N/A
Above Flood Level: Yes: No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	60-90	Sheet 2	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	20-80	20-80	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject To Spray	-	-	-	-	
Radiation (Rad)	4×10^4	Sheet 2	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Subject To Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Report F-C4124 performance qualification tests of four valve motor operators.	

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Sheet 2

Component ___Limitorque Valve Actuator MO-7066___

This Limitorque actuator contains a Peerless ac motor with Class B insulation. It is located outside containment in the post-incident room. The valve is actuated manually in the control room when recirculation of the containment sump water begins. The valve opens the cooling water flow path for the post-incident heat exchanger cooling from the fire water system.

The valve, being in the post-incident room, does not experience a harsh environment until after recirculation begins and the room heats up and radiation shine from the sump water is present. Because the valve is operated while the area remains at normal conditions, it is qualified for operation. Once opened, it will not be necessary to close the valve. In the event it cannot be opened remotely, this valve, or a bypass, can be operated manually in the post-incident room.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
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Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Main Steam	Operating Time	30 Days (Note 2)	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: MO-7067	Temperature (°F)	130	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Valve Actuator; Manufacturer: Limitorque	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: SMB-1 (Note 1); Purchase Order Number:	Relative Humidity (%)	20-80	20-80	Section II.E	Sheet 2	Evaluation	
	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Turbine Bypass Isolation Valve	Radiation (Rad)	1.6×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Turbine Pipe Tunnel; Elevation:	Submergence	Not Subject To Submergence	-	Section II.E	-	-	
Flood Level Elevation: N/A; Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. Nuclear Power Station Qualification Type Test Report Limitorque Valve Actuators for BWR Service Project No 600376A.	1. Peerless Class B insulated motor. 2. This valve is not required for HELB outside containment.

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: Big Rock Point
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Component Sheet No.:
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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Limitorque Valve Actuator Qualification for Nuclear Power Station Service, Report B0058. 3. Qualification Type Test Report Limitorque DC Valve Actuators for Nuclear Power Station Service Conditions, Report No B0009. 4. Qualification Type Test Report Limitorque Valve Actuators for Class IE Service Outside Primary Containment, Report No B0003. 5. Nuclear Engineering Handbook, H Etherington, et al, Section 10, Table 5.	

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Sheet 2

Component Limitorque Valve Actuator MO-7067

This valve is one of several valves that provide backup isolation for the MSIV. No credit for these valves, other than the MSIV, has ever been taken in any safety analysis. The valve is located outside containment in the turbine building pipe tunnel area. It will not be required to operate in the event of an HELB outside containment. The valve is closed on a containment isolation signal on high containment pressure and low reactor water level. Assuming the break is small and operator action is required to close the valve, 10 minutes are assumed to be the time for the operator to actuate the isolation valve switch. The emergency procedure immediate action is to perform this operation. Following closure of the valve, it will remain closed for the duration of the post-LOCA period.

The valve in its normal and post-LOCA position will be exposed to atmospheric pressure and normal humidity. Temperatures in the area range above normal pipe tunnel temperatures due to the adjacent main steam line. They will remain above normal in the 10 minutes it takes to operate the valve. The temperature will then decay to ambient.

Tests by Limitorque of several SMB type actuators more than encompass the post-LOCA outside containment environment this valve will be required to operate in. Limitorque states in report B0058 the SMB family of actuators is generic in that all actuators use the same materials and the only difference is the size of the actuator. Franklin Test Report No F-C3441 of a Limitorque SMB actuator, with Class RH insulated motor, subjected it to a temperature of 340°F and pressure of 108 psig in saturated steam conditions. The unit was preaged for 100 hours at 180°C, which by the 10°C rule, is 48 years at 60°C (140°F). The unit was also pre-irradiated to 204 megarads. This test report provides the basis for qualification of the actuator. The actuator motor, a Peerless Class B insulated dc-operated, has not been type tested in the Limitorque actuator assembly.

The outside containment conditions will not be as harsh as those of the tested actuator. The motor, in fact, will only be subject to its normal high-temperature condition in the time until it is required to operate. Radiation at 1.6×10^5 rads for its 40-year (1.4×10^5 rads) plus LOCA (2×10^4 rads) is within the threshold limit for motor materials listed in the Nuclear Engineering Handbook, H Etherington, et al. Limitorque tests of a dc Class H insulated motor and ac Class B insulated motor, in Test Reports B0009 and B0003, irradiated the assemblies to 2×10^7 and 1×10^7 rads, respectively. The dc motor was also aged at 180°C for 100 hours.

Since the effects of thermal and radiation aging on this specific dc motor are undetermined, the motor will be replaced or age qualified by June 30, 1982. The SMB actuator is determined to be qualified. As stated earlier, no credit is taken for this valve in any safety analysis; therefore, no immediate action is required.

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

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Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Test and Evaluation	
Plant I.D. Number: MO-7068	Temperature (°F)	235	235	Section II.E	1	Test and Evaluation	
Component: Valve Actuator; Manufacturer: Limitorque	Pressure (PSIA)	41.7	41.7	Section II.E	1	Test and Evaluation	
Model Number: SMA-00	Relative Humidity (%)	100	100	Section II.E	1	Test and Evaluation	
Purchase Order Number:	Spray	Lake Water	Tap Water	Section II.E	1	Test and Evaluation	
Function/Service: Containment Spray Valve	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Test and Evaluation	
Location: Containment; Elevation: 625	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. FIRL Report F-C4124 performance qualification tests of four valve motor operators.	

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Sheet 2

Component Limitorque Valve Actuator MO-7068

This Limitorque actuator contains a Peerless ac motor with Class B insulation. Presently the valve is not required to operate unless a failure occurs in MO-7064. Operation of both containment sprays would be detrimental to the core spray flow distribution and, therefore, MO-7068 is closed with its breaker pulled. To operate, it would therefore require manual closing of the breaker and manual operation of the controller. If needed, the valve would be used intermittently until the containment post-LOCA environment returned to ambient. From the LOCA envelope, this would take approximately one day. From this period on, the valve will remain closed.

A LOCA simulation test report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators," subjected this valve to a 36-hour test which followed the BRP LOCA envelope for this period of time. The test included 24 hours continuous spraying and intermittent spray for 1-hour periods at hours 27, 31 and 35. Relative humidity was maintained at or near 100% by (1) use of saturated steam to obtain initial temperature and pressure rise to 240°F, (2) use of fine mist water over the test specimen, (3) use of saturated steam injections to maintain chamber temperatures.

Radiation and thermal aging qualification testing has not been done for this type actuator. Generally, most component materials used in the manufacturing of actuators and motors can withstand a threshold damage limit of at least 4.0 E06 rads. the effects of thermal and radiation aging are, however, unknown. Based on the LOCA simulation test results though it is expected this actuator will operate for the required one-day period without significant degradation due to aging. To meet the guideline requirements, the actuator assembly will be replaced or rebuilt and qualified by June 30, 1982.

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Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Test and Evaluation	
Plant I.D. Number: MO-7070 MO-7071	Temperature (°F)	235	Sheet 2	Section II.E	1	Test and Evaluation	
Component: Valve Actuator	Pressure (PSIA)	41.7	Sheet 2	Section II.E	1	Test and Evaluation	
Manufacturer: Rotork	Relative Humidity (%)	100	Sheet 2	Section II.E	1	Test and Evaluation	
Model Number: 14 A Syncroset	Spray	Lake Water	Sheet 2	Section II.E	1	Test and Evaluation	
Purchase Order Number: S/N S3605 S3606	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Test and Evaluation	
Function/Service: Backup Core Spray Valves	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Test and Evaluation	
Accuracy: Spec: Demo:	Submergence	Not Subject To Submergence	-	-	-	-	
Location: Containment							
Elevation: 637							
Flood Level Elevation 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Report F-~4124 performance qualification tests of four va~ motor operators. 2. Rotork ~t Report No TR422 - Heat Aging Class B, NA2 motor for outside containment safety-related duty.	

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: Big Rock Point
Docket: 50-155

Component Sheet No.:
Revision:
Date:

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Letter Rotork, RHA to CP Co, JLK dated October 17, 1980 with attached summary for "A" range actuators. 4. Rotork Test No N/14/2, "Actuator Radiation Subjection." 5. Limitorque Qualification Test Report No B0003.	

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Sheet 2

Component Rotork Valve Actuator MO-7070, -7071

These Rotork actuators have ac motors with Class B insulation and weatherproof enclosures. MO-7070 and MO-7071, the backup core spray valves, were initially installed in the Plant in 1970. The core spray valves are actuated on low reactor water level coincident with reactor pressure less than 200 psig.

Report F-C4124 "Performance Qualification Tests of Four Valve Motor Operators" subjected the MO-7070 to 50 minutes of LOCA simulation at 240°F and 43.7 psig with tap water spray and relative humidity of 100%. Humidity was maintained at or near 100% by the following means: (1) use of saturated steam to obtain the initial temperature and pressure rise to 240°F, (2) use of fine mist water over the specimens, (3) use of saturated steam ejections to maintain chamber temperatures. MO-7070 failed the first test as a result of an electrical short across a preexisting damaged terminal strip. The terminal strip was replaced with in-line insulated splices and successfully retested under the same environmental conditions. The like actuators in the Plant were modified with the in-line splices. The motor of the tested Unit MO-7070 was replaced with a new unit in 1979 after the original failed due to unknown reasons.

The backup core spray valves, like the primary core spray valves (MO-7051 and -7061), operate within one hour maximum following a break. They may be required to close after recirculation has begun (maximum of 21 hours). In such a case, the primary spray line would provide core cooling.

Rotork has conducted testing for aging qualification. Rotork Technical Report TR 422 subjected an NA type actuator with a Class B insulated motor to an aging test which simulated 48 years at 60°C. The unit in the test was placed in a furnace for 100 hours at 180°C. Forty-eight years at 60°C was determined by the 10°C rule. TR 422 did not include a LOCA simulation and although an NA type actuator w used in the test, the only difference between the NA and 14A unit, according to Rotork, is the motor housing material, cast iron versus cast aluminum. The results of the test, Rotork states, and the electrical characteristics will be exactly the same for the 14A as the NA type units. Rotork test No N/14/2 used a prototype unit built for material evaluation of the standard "A" and NA1 components. The actuator for MO-7070 and MO-7071 Type 14A are standard "A" units. The test results showed the "A" range components capable of withstanding 30 megarads during their 40-year life, according to Rotork. The test unit, however, was equipped with a Class H insulated motor. Limitorque actuators (Report B0003) with Class B motors have been irradiated to 20 megarads and have been subjected to outside containment HELB conditions for a 16-day period and have operated satisfactorily. Since the valves operate within one day and remain in position, open or closed, thereafter the 30-day radiation dose of 7.3×10^5 affect the aging of the components after it has completed its required function. The one-day dose of 4.9×10^5 is conservatively what the valve will be exposed to during its operating time. This dose is within the qualification dose of most materials.

In summary, based on the testing of the MO-7070 actuator and the tests of Rotork showing aging and radiation qualification, the units are considered

Sheet 2 (Contd)

satisfactory for their intended service. Further, the operating time is less than one day, after which time the units will remain in position and not be required to actuate. Their motor starters located in the station power room, a nonharsh area, will not misoperate due to the environment they are in. The actuators are considered acceptable for use.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION
System: Post-Incident
Plant I.D. Number: MO-7072
Component: Valve Actuator
Manufacturer: Rotork
Model Number: 14 A Syncroset
Purchase Order Number: S/N S3603
Function/Service: Core Spray Heat Exchanger
Accuracy: Spec: Demo:
Location: Core Spray Room
Elevation: 588
Flood Level Elevation: N/A
Above Flood Level: Yes: No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	60-90	Sheet 2	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	80	80	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject To Spray	-	-	-	-	
Radiation (Rad)	1×10^2	Sheet 2	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Subject To Submergence	-	-	-	Evaluation	

DOCUMENTATION REFERENCES	NOTES
1. Report F-C4124 performance qualification tests of four valve motor operators.	

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Sheet 2

Component   Rotork Valve Actuator MO-7072

This Rotork actuator contains a dc motor with Class B insulation and weatherproof enclosure. It is located outside containment in the post-incident room. The valve is actuated manually in the control room only in the event of a failure in the fire header system supplying core and containment spray water. It is only an emergency source of water and will not be normally used. In the event it is opened, it will be closed within 21 hours when recirculation begins. It will not have to operate in a harsh environment and will be maintained closed throughout the incident.

Because the valve operates prior to the area experiencing higher than normal temperatures and radiation doses, the actuators are qualified for operation. Even in the event the valve actuator will not operate remotely, the valve can, if necessary, be operated manually in the post-incident room prior to recirculation.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number:	Temperature (°F)	152	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Motor Starter Manufacturer: GE	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: CR 109	Relative Humidity (%)	20-80	20-80	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Motor Starter for MO-7066 to Core Spray Hx	Radiation (Rad)	4×10^4	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Post-Incident Room Elevation:	Submergence	Not Subject to Submergence	-	-	-	-	
Flood Level Elevation N/A Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES

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Sheet 2

Component GE Motor Starter

This motor starter is for MO-7066, the cooling water valve to the core spray heat exchanger. The valve is actuated manually in the control room when the containment sump recirculation mode begins. The time required to reach recirculation level in the sump is a maximum of 21 hours. During this time, the core spray heat exchanger room remains at ambient conditions with respect to temperature, humidity, pressure and radiation. Temperature and radiation following the start of the recirculation are increased due to the temperature and radiation level of the sump water and temperature from the core spray pump motor heat. After the valve is required to open to allow cooling water to the heat exchanger, it will remain open for the duration of the event. If the valve actuator of the subject starter fails to operate to open the valve, it or its bypass can be opened manually in the core spray room. The time to begin recirculation is from 4 to 21 hours. No feasible failure of the starter can be determined that would cause the valve to reclose once it has been opened.

Because the starter operates when the room is at ambient conditions, and failure prior to this can be overridden by manual action, the starter is considered acceptable.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION

System: Post-Incident

Plant I.D. Number: P2A, P2B

Component: Core Spray Pump Motor

Manufacturer:

Model Number: General Electric

Purchase Order Number: SK4364XJ1A11

Function/Service: Supplies Core Spray Water

Accuracy: Spec:
Demo:

Location: Core Spray Room

Elevation:

Flood Level Elevation: N/A
Above Flood Level:
Yes: No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	152F	194	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	80	80	Section II.E	Sheet 2	Evaluation	
Spray	None	–	–	–	–	
Radiation (Rad)	4×10^4	Sheet 2	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Section I[illegible] & Sheet 2	Evaluation	
Submergence	None	–	–	–	–	

DOCUMENTATION REFERENCES	NOTES
1. Nameplate data taken from motor at plant. 2. Letter R Marusich, Consumers Power Company, to Don Hueling, General Electric, verifying telephone conversation concerning motor data.	

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Sheet 2

Component Core Spray Pumps

The core spray pumps are located in the core spray pump room which is heavily shielded from containment. The room contains normal environment except during recirculation phase of the LOCA. During this time, the room reaches 152°F in 400 hours and remains at that level throughout the remainder of the accident. The room temperature is obtained from a calculation which uses motor heat loss and recirculation piping heat loss as heat input to the room. The radiation results from the recirculation of radioactive water from the sump, through the core spray heat exchanger and back to the core. All common motor materials can withstand this radiation level. The pumps are started periodically but have never run for any length of time. The manufacturer states that the motors have a life of 10-years' continuous operation. Since the motors are not used except for LOCA and the fact that they are Class B insulation, leads to the conclusion that they have sufficient life. The motors have a 40°C temperature rise (nameplate data). Class B insulation is good to 130°C so that the room ambient can reach 90°C without harm to the motors. This is greater than the temperature which the room will reach. The lubricating oil is from AMOCO and is American Industrial Oil Number ISO-VG #32. The oil is checked each year and changed when needed. The oil will be changed this refueling outage (1980) to assure adequate life. The oil will be changed periodically thereafter per manufacturer's instructions. The lubricating oil is considered to be able to withstand the radiation as the levels are so low. Due to the above, the motors are considered qualified.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	21 Hours	21 Hours	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: PDIS-7814 PS-638	Temperature (°F)	See Sheet 2	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Pressure Switch; Manufacturer: ITT Barton Mercoid	Pressure (PSIA)	See Sheet 2	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: Barton - 289A Mercoid - DAW-23-153	Relative Humidity (%)	See Sheet 2	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	None	-	-	-	-	
Function/Service: Barton - Basket Strainer Hi dp Alarm; Mercoid - Core Spray Pump Discharge Press	Radiation (Rad)	See Sheet 2	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	See Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Core Spray Room; Elevation:	Submergence	None	-	-	-	-	
Flood Level Elevation: N/A; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
	1. PDIS-7814, ITT Barton Model 289A; alarm for high differential pressure across PIS strainer PS-638, Mercoid Model DAW-23-153 (from plant data); alarm for core spray pump start failure.

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Sheet 2

Component Pressure Switches PS-638 and PDIS-7814

Both of these switches are in the core spray room. This room is not subjected to harsh environments from the LOCA until the containment sump is filled and the operator switches to the recirculation mode for core cooling water. Pressure switch PS-638 senses pressure on the discharge header from the core spray pumps. If the pressure is < 50 psig, an alarm is provided in the control room. Its primary function is to inform the operator of the failure of a core spray pump to start and therefore is not required subsequent to the start of the recirculation mode. Core spray flow indication is also provided by FT-2162 and FT-2163.

Differential Pressure Switch PDIS-7814 measures the dp across a basket strainer that filters firewater to the core sprays. This is an emergency path for core spray water addition, and would not normally be used (Reference MO-7072, Sheet 2). Until recirculation, the environment is normal. The operator is directed by procedure to align the basket strainer to the "open basket" position prior to going to the recirculation mode. In the "open basket" position, PDIS-7814 serves no purpose.

Thermal aging is not considered to affect LOCA qualification of these devices for the following reasons: (1) The normal temperature is 70°F, a low temperature in terms of significant thermal aging, (2) the function of these devices is completed prior to being subjected to a harsh environment and (3) the switches have backup instruments that provide the necessary information, that is FT-2162 and FT-2163.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	1 Hour	1 Hour	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: PS-7064A, PS-7064B	Temperature (°F)	Sheet 2	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Pressure Switch Manufacturer: Static-O-Ring	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: 12L-AA5-FSS	Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	None	–	–	–	–	
Function/Service: Starts Enclosure Spray Timer	Radiation (Rad)	2×10^4	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + 1 Hr LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Electrical Penetration Room Elevation:	Submergence	None	–	–	–	–	
Flood Level Elevation N/A Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. Static-O-Ring Catalog, Revision 4, 1972.	

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EQUIPMENT DESCRIPTION

System: Reactor Protection

Plant I.D. Number: PS-664 Through 667

Component: Previous Switches

Manufacturer: Static-O-Ring

Model Number: 4NN-E411-YX5TT

Purchase Order Number:

Function/Service: Reactor Scram on High Containment Pressure

Accuracy: Spec: Demo:

Location: Electrical Penetration Room

Elevation:

Flood Level Elevation: N/A

Above Flood Level: Yes: No:

Parameter	Environment: Accident	Environment: Qualification	Documentation References: Accident	Documentation References: Qual	Qualification Method	Outstanding Items
Operating Time	1 Hour	1 Hour	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	See Sheet 2	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	Atmospheric	Atmospheric	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Spray	None	-	-	-	-	
Radiation (Rad)	2×10^4	See Sheet 2	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + 1 Hr LOCA	See Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	None	-	-	-	Evaluation	

DOCUMENTATION REFERENCES	NOTES
1. Static-O-Ring Catalog, Revision 4, 1972.	

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Component Pressure Switches PS-7064A, B and PS-664 Through PS-667

These switches are located in the electrical penetration room. The room is adjacent to the containment and has a common wall (the shell) with it. PS-664 through -667 are part of the reactor protection system causing a scram on high containment pressure. Once it actuates, the signal is sealed in. PS-7064A and B actuate on high containment pressure also. Actuation of this switch starts the timer in the enclosure spray valve opening circuitry. Once actuated, the signal is sealed in. As a result, the switches will only experience the environment, which is transferred through the sphere, due to the containment atmosphere at the high containment trip set point (1.7 psig). The highest containment wall temperature at the trip set point is about 150°F. This results from a very small steam line break. Larger breaks result in faster time of high containment pressure trip and lower wall temperatures. The temperature at the switches due to this break is less than this due to thermal inertia in the containment wall, atmosphere of the cable penetration room and switch housing. Since the temperature inside the switch is so low, it is considered to be qualified for temperature. The switches are enclosed in an explosionproof housing or waterproof housing and are not affected by humidity. The normal environment in the room is 40°F-100°F (seasonal variation). Since the conditions at time of actuation are not much different from normal, no significant stress will be placed on the switches during the event. Therefore, the switches are likely qualified for aging. For further arguments, see "Aging" in the body of the report.

The radiation levels are based on 100% core melt at t=0 integrated to one hour after the start of the event. For the large break LOCAs, the core uncovers early in the event, but high containment pressure also occurs very early (< 1 hour) so that the radiation levels will be much less. For the small main steam line break upon which the 1-hour operational time is based, the core does not uncover and 100% core melt will not occur. Even if forced to assume 100% core melt at t=0 for a small main steam line break, the radiation levels are such that all common materials used in the construction of these switches can withstand them. Therefore, the switches are qualified for radiation.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	1 Hour	1 Hour	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: PS-IG11A Thru IG11H	Temperature (°F)	235	260	Section II.E	6, 1, 4, 5	Test	
Component: Pressure Switch	Pressure (PSIA)	41.7	See Sheet 2	Section II.E	6, 1, 4, 5	Test and Evaluation	
Manufacturer: Static-O-Ring	Relative Humidity (%)	100	100	Section II.E	6, 1, 4, 5	Test	
Model Number: 9TA-S4-11SSX12	Spray	None	-	-	-	-	
Purchase Order Number:	Radiation (Rad)	$<2 \times 10^5$	$>10^6$	Section II.E	2, 3	Evaluation	
Function/Service: Provide Low Reactor Pressure Permissive To Open Core Spray Valves	Aging	40 Years + 1 Hr LOCA	See Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo: Location: Containment Elevation: 590 and 621	Submergence	No	-	-	-	-	
Flood Level Elevation: 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Letter M J Aklinski MEA Inc to D L Giuliani, F Conroy Mechanical Contractors November 10, 1972 entitled Environmental Test for Pressure Switches.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. BRP Task Force File 199 H J Palmer to R B Cherba May 23, 1975. 3. Nuclear Engineering Handbook, Etherington 4. D J Blaies Microswitch to G Blickley Mechni Arts Assoc May 16, 1975. 5. Static-O-Ring Pressure Switch Company Certificate of Compliance 7101-156. 6. Static-O-Ring Catalog Rev 4-72.	

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Sheet 2

Component Static-O-Ring Pressure Switch PS-IG11A-H

These pressure switches are used to provide a core spray valve open signal when reactor pressure is ≤ 200 psig. In the electrical scheme, these switches have contacts in series with the reactor water level switches (LS-RE09). Therefore, both these pressure switches and the reactor water level switches must remain operable for the same length of time. The time required was established at 1 hour. The basis for this time is that it is the longest time for automatic actuation of the core spray system when the LOCA is a small steam line break (0.008 ft^2). For the 0.008 ft^2 break, the reactor depressurizing system (RDS) must first operate to reduce the reactor pressure to ≤ 200 psig. The RDS will operate when low reactor water level is reached in this small steam line break senario.

Per the Loss of Coolant, Emergency Procedure EMP 3.3, the operator is directed to manually initiate the RDS if he deems it necessary. With the loss of steam drum level indication and alarm as well as all of the other symptomatic indications, it is likely that the operator would take the required action early in the accident rather than wait an hour for automatic action. Therefore, in reality, the instruments would not have to endure an hour in the environment caused by a small steam line break. Further, the small steam line break produces a less severe environment than the large LOCAs until the RDS functions. The larger sized LOCAs allow the instruments to perform their functions early in the events, thus significantly reducing the lifetime required to demonstrate qualification.

The actual atmospheric conditions under which these switches will operate are pressure less than 39 psia and temperature less than 235°F.

These switches were environmentally qualified in a test. During this test, the pressure switches were placed in a test chamber and nitrogen was added. The conditions were raised to 260°F and 20 psig and held for 2 hours. This is of a longer duration than required. The switches were also test-operated during this time.

The test temperature was greater than that needed for qualification but the pressure was 7 psig lower. The difference in pressure is not significant because the switches are in an explosion proof housing as compared to a NEMA 4 housing used on the test specimen. The additional pressure will have no effect on the explosionproof housing.

Steam and spray were not used in the test; however, the sealed explosionproof housing will prevent this from having any significant effect on the switches. Therefore, the lack of use of steam or sprays does not invalidate the test.

There were differences between the test specimen and the installed switch but these differences were not significant and do not invalidate the test. (1) The pressure range was different. (2) The test specimen had a weathertight housing vs a sealed explosionproof housing for the installed specimen. (3) The microswitch used in the installed specimen was evaluated by

Sheet 2 (Contd)

the manufacturer to be able to withstand the environment (Ref 4). (4) The tested pressure switch had 5 options which were not listed. The installed transmitter has 12 options. The 12 options make the switch more insensitive to the environment (Viton O-ring cover gasket, silicone diaphragm, high temperature shaft, 316 SS primary diaphragm, Viton O-ring) and also include items like certification, data sheets, cleaning and packaging. It is therefore considered that the options in the tested sample which may not be in the installed switch are not important.

The switch's ability to withstand radiation was evaluated. It was found that even the most sensitive component can withstand greater than 10^6 rads.

The pressure switches actuate early in the event and the LOCA event is expected to thermally age the components very little. The normal aging environment is a temperature averaging 70°F. Since these switches are on a periodic surveillance for calibration, a failure could be detected. No failures have been found. It is concluded that (1) with the uncertainty in determining the lifetimes for age sensitive materials (see Section IV), (2) with the low ambient temperatures experienced by the instruments, (3) with the short time needed during an accident and (4) with the known good history of the instruments during calibration intervals. These instruments are acceptable for continued operation.

The switches are encased in a sealed explosionproof housing. This results in low temperature increases to the switch intervals prior to actuation.

The pressure switches are qualified for the environment they will experience. They were tested for high temperature and pressure for a duration longer than the time required for operation in the event of an accident. The test did not include sprays or humidity; however, the switches are sealed in an explosionproof housing and will not be affected by spray or humidity. They were evaluated and found able to withstand the radiation and aging.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Post-incident	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: PT-173, 174, 187	Temperature (°F)	182	220	Section II.E	1	Evaluation	
Component: Pressure Transmitter	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Manufacturer: Rosemount	Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Model Number: 1151GP	Spray	None	-	-	-	-	
Purchase Order Number:	Radiation (Rad)	5.2×10^4	Sheet 2	Section II.E	Sheet 2	Evaluation	
Function/Service: Containment Pressure Indication and Vacuum Relief	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Submergence	None	-	-	-	-	
Location: Electrical Penetration Room							
Elevation: 599							
Flood Level Elevation: N/A							
Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Rosemount Product Data Sheet 2260.	

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Sheet 2

Component Rosemount 1151GP Pressure Transmitters

These pressure transmitters are used to indicate containment pressure and, in the case of PT-173 and PT-187, actuate valves for vacuum relief. The vacuum relief function, if needed, will be required up to 12 hours after the LOCA. The transmitters are located in the electrical penetration room. The room experiences a temperature rise during the LOCA as it is adjacent to the containment (spherical steel structure). To determine the temperature in this room, a CONTEMPT run was performed for the double-ended rupture of the main steam line. This results in the highest containment temperature. One of the assumptions was that the containment is completely insulated and, therefore, no heat is transferred out of containment. This is a very conservative assumption. The peak inside containment wall temperature was found to be 182°F. The room temperature during the accident will be lower than this, due to the conservative assumption of zero heat transfer out of containment and the time lag to transfer heat through the shell and through the room to the transmitters (located at least 6 feet away). The transmitters are able to operate in ambient conditions of 220°F and, therefore, are qualified for temperature.

The radiation dose, which these transmitters will obtain, is calculated to be 5×10^4 rads in 30 days. The containment pressure is back to atmospheric in approximately 24 hours. The dose in 24 hours is 3×10^4 rads (ratio of containment atmosphere dose at 24 hours to that at 30 days). The sensitive parts of the transmitter are housed in an explosion proof casing and are also shielded by other components. The dose is, therefore, even lower. All commonly used materials can withstand this radiation level and it is concluded that the transmitter is qualified for radiation.

No data on aging could be located. The transmitters are in a normal environment which ranges from 40°F to 100°F (seasonal variation). CP Co will continue to search for age sensitive materials and, if found, will be replaced by June 1982. Continued operation is justified by the arguments in Section IV.

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EQUIPMENT DESCRIPTION
System: Reactor Vessel - General
Plant I.D. Number: PT-IA07C
Component: Pressure Transmitter
Manufacturer: Rosemount
Model Number: 1152GP 9A92PB
Purchase Order Number:
Function/Service: Reactor Vessel Pressure
Accuracy: Spec: Demo:
Location: Containment
Elevation: 597'-0"
Flood Level Elevation: 590'-0" Above Flood Level: Yes: X No:

Parameter	ENVIRONMENT Accident	Qualification	DOCUMENTATION REFERENCES Accident	Qual	QUALIFICATION METHOD	OUTSTANDING ITEMS
Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	235	316	Section II.E	1, 2	Test	
Pressure (PSIA)	41.7	84.7	Section II.E	1, 2	Test	
Relative Humidity (%)	100	100	Section II.E	1, 2	Test	
Spray	Lake Michigan Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Radiation (Rad)	7.3×10^5	5×10^6	Section II.E	1, 2	Test	
Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Subject to Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Quality Certification of Compliance Data Sheet, November 12, 1976. 2. Qualification Tests for Rosemount Pressure Transmitter Model 1152, Report 117415.	

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Sheet 2

Component Rosemount 1152GP Pressure Transmitter

Reference 1 stated that the installed transmitter meets the nuclear qualification data with respect to radiation, vibration, temperature and pressure as verified in Rosemount Report 117415 (Reference 2). The pressure transmitter tested in this report was 1152DP4A22. The transmitter first underwent thermal aging. The thermal aging test was two temperature cycles of 100°F to 0°F to 200°F to 100°F. All temperatures were sustained for a minimum of one hour. After each hour of stabilization, readings were taken. This was done in accordance with IEEE-323 (1971). Next, the transmitter was exposed to 1×10^6 rad/h for five hours. Data was taken at each hour. Next, the unit was seismically tested and then environmentally tested. The test sequence was (1) to admit hot air to raise the conditions in the test chamber to 350°F and 60 psig for 10 minutes, (2) admission of saturated steam raising the temperature and pressure to 316°F and 70 psig holding for 1 hour, (3) to reduce to 303°F and 55.4 psig and holding for 7 hours and (4) to reduce to 230°F and 6 psig and holding for 42 hours. These environmental conditions exceed that required to demonstrate qualification. The test duration is a total of 50 hours which does not cover the time at which ambient conditions are reached at Big Rock Point (72 hours); however, the duration is considered sufficient because the test conditions greatly exceed the environmental conditions following a LOCA at Big Rock Point. The radiation levels are in excess of those required to demonstrate qualification. The transmitters are encased in a housing so that the additional dose due to beta radiation is negligible. The test did not include spray. The spray at Big Rock Point consists of Lake Michigan water. Since this transmitter is enclosed and passed the radiation and environmental tests, sprays will not produce any adverse effects. Further, this transmitter is located in a room such that it will not be subjected to containment sprays directly. No further aging information could be located other than the aging test presented. This test, coupled with the fact that the transmitters are in a normal environment of 50°F to 90°F (seasonal variation) and the arguments presented in Section IV, provides qualification of aging.

The transmitters underwent an aging test, then a radiation test and then an environmental test and met the acceptance criteria placed on the operation both during and after each test. The transmitters are enclosed and, therefore, not affected by the spray. Also, they are shown to be qualified for aging. Therefore, these transmitters are considered to be fully qualified.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radiation Indication	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: -	Temperature (°F)	182	Sheet 2	Section II.E	Sheet 2 & Ref 2	Evaluation	
Com[illegible]: Radia[illegible] Monitor	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2 & Ref 2	Evaluation	
Manufacturer: Technical Assoc	Relative Humidity (%)	100	100	Section II.E	Sheet 2 & Ref 2	Evaluation	
Model Number: CP-MU	Spray	None	-	-	-	-	
[illegible]chase Order Number:	Radiation (Rad)	5.2×10^5	1×10^8	Section II.E	1, 2, 3	Evaluation	
Function/Service: Failed Fuel Monitor	Aging	Sheet 2	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Submergence	None	-	-	-	-	
Location: Electrical Penetration Room							
Elevation:							
Flood Level Elevation N/A							
Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. Effects of Radiation on Materials and Components, Kircher and Bowman.	

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Technical Associates High Range Survey Meter Description and Specifications. 3. Nuclear Engineering Handbook, Etherington.	

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Sheet 2

Component Survey Meter - Core Damage Monitor

The survey meter is located in the electrical penetration room. Its function is to detect radiation inside containment and provide this indication to the operator. The meter is not safety-related and plant shutdown can be achieved without its use. The probe consists of an ionization chamber which is encased in an aluminum housing. The unit is designed to operate in high humidity. The electronics are in a separate room which is a "mild" area. The cable used to connect the probe to the electronics is jacketed and has polyethylene insulation.

The meter was installed late in 1979 and is temporary until the high-range monitors are installed as part of the TMI-related modifications. The probe itself is an ion chamber. Due to its construction, it can withstand 182°F. Failure of this probe will not affect plant shutdown. The probe is qualified for aging as it is brand new and the temperatures in the room over the 30 days will not exceed 182°F. The temperatures in the room will most likely not even reach 182°F due to thermal inertia in the containment shell, the room atmosphere and the housing and cable jacket. The monitor is qualified for the temperature and radiation as the electronics are placed in another room. The temperature and radiation levels of 5×10^{6} will not affect the aluminum housing or polyethylene insulation (1×10^{8}). Therefore, the monitor is qualified.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radwaste	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4876	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: HTX-8300C61RF	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Fuel Pit and Reactor Drain Isolation Valve	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment Rm 424; Elevation: 591	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation: 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. ASCO Engineering Report No 47 for Catalog No 8300C61RF.	

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radwaste	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4869, 4891	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: 830060RF	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	–	–	–	–	
Function/Service: Containment Clean and Dirty Sump Isolation Valve	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment Rm 424; Elevation: 591	Submergence	Not Subject To Submergence	–	–	–	–	
Flood Level Elevation: 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES

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Sheet 2

Component ASCO Solenoid Valves SV-4876, 4869, 4891

These solenoid valves are automatically de-energized by the reactor safety system on an isolation trip from either the containment high-pressure or low-reactor water level signals. The valves are supplied with redundant valves outside containment which are also tripped by the same signals. The valves outside containment will not be subject to a harsh environment during a LOCA inside containment except for higher than normal radiation exposure. Once these valves are de-energized by the isolation trip, they remain in this mode. Immediate operator action in the emergency procedure is to actuate the isolation switch to close these valves. This action would back up the automatic trip or provide a trip in the event one hadn't been initiated. Since the valves are tripped automatically on a high containment pressure signal at 1.5 psig, or manually prior to this, they will not have to operate in as severe an environment as described on the qualification sheet. ASCO has stated the peak temperature and pressure of 235°F and 41.7 psia will probably not have any effect on the operability of the valves. From Figures 1 and 2 when the pressure reaches 1.5 psig, temperature is below 140°F. Relative humidity of 100% may cause shorting of the solenoid according to ASCO. In this event, the result will be de-energizing of the solenoid and closing of the control valve. It will fail safe. The valves are located in areas that are protected from the containment sprays but will have operated prior to the spray initiation which has a higher containment pressure set point. The 2-hour radiation exposure of 2.0×10^5 rads was the dose that could be received at the center line of containment. This dose can be halved as the valves are mounted on large concrete walls. Conservatively using 10^5 rads, the valve materials of BUNA N and Zytel 103 have a threshold limit of greater than 10^6 rads which qualifies the valves to operate with the radiation exposure they receive. The 30-day exposure of radiation to the valves of 7.3×10^5 can also be halved and is also below the threshold limit of the valve materials. The materials will not degrade to failure with the valves in their de-energized mode during the post-LOCA period, due to the radiation exposure.

Both the SV-4869 and 4891 have, within the past month, had replacement spare parts installed in them. The replacement parts include the BUNA N O-ring and seat which is also the elastomer used in SV-4876 along with Zytel 103. SV-4876 is not presently on a PM replacement program. It, however, has a metal seat and Zytel disc with O-rings of BUNA N. Zytel, in addition to having a radiation resistance of 5×10^6 rads, is also a high temperature material (Reference 1) which has been used in the scram solenoids; and, in one test, endured 204 hours at a maximum temperature of 415°F and 250,000 operations. The material has not been shown to be susceptible to aging failures.

The valves are considered acceptable for present use because of their ability to withstand LOCA temperature, pressure and radiation. Replacement of parts and age sensitive materials also provide acceptability. There are also redundant valves located outside containment which de-energize on an isolation signal and remain in this mode. According to ASCO, failure of the materials with the valve de-energized will leave the valve in its fail-safe position.

Sheet 2 (Contd)

The effects of humidity will also leave the valve in its fail-safe position. The valves, however, do not meet the qualification guidelines as type testing of these valves has not been done. Therefore, the valves will be replaced by June 30, 1982.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Cleanup	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4879 Component: Solenoid Valve	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Manufacturer: ASCO	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: 831620 (Note 1)	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Resin Sluice Containment Isolation Valve	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment Elevation: 613	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
	1. Equipped with HVA-90-441-1A pilot head.

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Sheet 2

Component ASCO Solenoid Valve SV-4879

This solenoid valve is normally de-energized with its three redundant resin sluice control valves closed during power operation. When de-energized, the solenoid and control valves are in their fail-safe mode. The control valves are only opened in transferring resins and are opened only for short periods of time. Resin transfer takes place on an average of once per year and complete transfer can be done in about four hours. The valves are closed on an isolation signal from either high containment pressure or low reactor water level. If open, the valve will be closed automatically within a maximum of one hour following a break. See Section II.D. For small breaks, manual action by an operator is assumed in 10 minutes. Automatic initiation of an isolation for large breaks will occur within seconds of a break. Immediate operator action following symptoms of a break is to actuate the isolation valve switch which will back up or initiate valve closure. A normally closed manual valve outside containment also provides a redundant isolation valve. In the improbable event a break occurs when the line is open, the automatic and redundant valves inside containment provide the containment boundary.

The pilot head in the solenoid is the same model used for the scram solenoid valves and was installed in 1978. It has a NEMA 4 watertight enclosure that will prevent the 100% humid atmosphere from affecting the performance of the valve. Since the automatic isolation signal will be initiated at 1.5 psig, or isolation may be manual prior to this, the valves will not have to operate in as severe an environment as described on the qualification sheet. From Figures 1 and 2 when pressure reaches 1.5 psig for all break sizes, temperature is less than 140°F. A temperature of 235°F and a pressure of 41.7, ASCO has stated, will not affect the ability to operate the solenoid. The valve elastomers are normally in ambient conditions inside containment whereas the scram solenoids of the same valve model normally are energized and are at a substantially higher temperature.

Aging deterioration from the containment temperature will be accelerated in the 3-day post-LOCA period in which the atmosphere returns to ambient. The operation of the solenoid will not be affected, however. With the present 5-year preventative maintenance program, the elastomers will remain qualified with respect to thermal aging.

The 30-day radiation dose of 7.3×10^5 rads for the radiation dose in air listed on the qualification report sheet can be halved due to the solenoid valve being mounted adjacent to a substantial cement wall. The BUNA N, Neoprene and Zytel 103 materials all have radiation damage threshold values of greater than 1.0×10^6 rads according to ASCO and other sources. Radiation aging will not be significant with the dose received and the preventative maintenance program will provide new components to maintain the qualification.

The solenoid valve is mounted in a metal cabinet which will protect it from any containment spray water.

Sheet 2 (Contd)

The valve is normally de-energized; and, if energized, will de-energize prior to the atmosphere becoming hostile, and will not be subject to any spray. It has a watertight enclosure which will prevent humidity from causing any shorting. The accident 30-day radiation dose is below threshold damage to its materials. It also has a periodic replacement program to replace aging susceptible materials. A manual valve outside containment also provides backup to the normally closed control valves. Based on these reasons, the valve, although not meeting guideline requirements, is considered acceptable as is.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radwaste	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4892 Component: Solenoid Valve	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Manufacturer: ASCO	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: 83006DR	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Treated Waste to Containment	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment Elevation: 604	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES

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Sheet 2

Component ASCO Solenoid Valve SV-4892

This solenoid valve isolates the treated waste to containment via manual operation from the control room. The control valve is normally closed with the solenoid de-energized during power operation. The control valve is in series with a check valve. Both valves are leak rate tested each refueling outage to verify the containment integrity. Manual operation would only be necessary if the control valve were opened to transfer water to the spent fuel pool system. In this event the check valve provides the containment boundary until the control valve is closed. The Plant Emergency Procedure will have to be modified to identify to the operator to close the control valve. After identification of a LOCA, closure of the valve would be expected within 10 minutes, which is the allowed operator action time. The action will certainly be done in the one-hour minimum operating time requirement given in the NRC guidelines.

The BUNA N materials used in the nonmetallic valve components will not be significantly affected by the LOCA temperature of 235°F during the time it takes to operate the valve. ASCO has stated this material is acceptable for continuous operation in ambient temperatures up to 180°F and up to 240°F for short periods of time. The LOCA pressure of 41.7 psia also should not affect the function of the valve. Similar solenoid valve enclosures routinely are subjected to higher pressures during containment integrated leak rate tests. ASCO has also stated that 100% relative humidity may cause the valve to short. This, however, will result in the valve de-energizing and closing the control valve. The valve is located in an area that is protected from the containment sprays and is mounted adjacent to a substantial concrete wall that will have the effect of halving the radiation dose. In the time it would take to operate the valve, less than 1.0×10^5 rads (2-hour gamma dose halved) would be absorbed. BUNA N has a threshold limit of 1.0×10^6 rads which is well above the 30-day gamma dose of 3.65×10^5 rads which would be conservatively assumed. Thermal and radiation aging does affect the BUNA N material; however, the valve is presently in a preventative maintenance replacement program to replace the aging susceptible components.

Although no qualification testing has been done on this model valve, the length of time required, insignificant effect of pressure and temperature on the operation, low radiation dose and because it is backed up by another valve which will fail passively, all provide reasons for its present acceptability. Furthermore, it is a normally closed valve. Failure due to shorting or aging deterioration, in conversations with ASCO, will result in a fail-safe operation; ie, the control valve will shut. Because the guidelines are not met, an acceptable replacement will be procured and installed by June 30, 1982.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radwaste	Operating Time	30 Days (Note 1)	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4895, 4896	Temperature (°F)	50-90	50-90	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: HTX-8300C61RF	Relative Humidity (%)	80	80	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Containment Clean and Dirty Sump Containment Isolation	Radiation (Rad)	7.6×10^4	7.0×10^6	Section II.E	1	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Turbine Bldg Pipe Tunnel; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation: N/A; Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. The Effects of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	1. Not required to operate for HELB outside containment.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Radwaste	Operating Time	30 Days (Note 1)	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4922	Temperature (°F)	50-90	50-90	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Manufacturer: ASCO	Relative Humidity (%)	80	80	Section II.E	Sheet 2	Evaluation	
Model Number: 830060R	Spray	Not Subject To Spray	-	-	-	-	
Purchase Order Number:	Radiation (Rad)	7.6×10^4	7.0×10^6	Section II.E	1	Evaluation	
Function/Service: Fuel Pit and Reactor Drain Containment Isolation Valve	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Submergence	Not Subject To Submergence	-	-	-	-	
Location: Turbine Bldg Pipe Tunnel							
Elevation:							
Flood Level Elevation: N/A							
Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. The Effects of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	1. Not required to operate for HELB outside containment.

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Sheet 2

Component ASCO Solenoid Valves SV-4895, 4896, 4922

These solenoid valves are automatically de-energized by the safety system on an isolation trip from either the containment high-pressure or low reactor water level signals. The valves are supplied with redundant valves inside containment (SV-4869, 4891 and 4876 with their respective control valves). Following de-energization, the valves will remain in the same state throughout the post-incident period. Failure of the elastomers in the solenoid valve will result in the control valve remaining in its fail-safe position; ie, closed. Automatic de-energization of the valve on an isolation signal will occur well within one hour (see SV-4876, 4869, 4891) after a break; and, because the valve is located outside containment, it will experience no harsh environmental changes in this time. Temperature, pressure and relative humidity will remain at ambient conditions outside of containment. The total integrated 30-day radiation exposure at the containment surface is 7.6×10^4 rads which is less than the threshold limit of greater than 1×10^6 rads for the valve elastomers.

The aging of the elastomers is not significantly accelerated during the post-LOCA period except, due to the higher than normal radiation which is shown above, the 30-day exposure is less than the threshold limit. The valves are not currently on a preventative maintenance program to replace the aged and worn components. In order to provide aging qualification, the valves will have the age susceptible components replaced by June 30, 1982.

In summary, the valves are not required for HELB outside containment. Their environment, except for radiation, is not harsh and the materials are resistant to the accident dose received. To provide qualification for aging, a preventative maintenance replacement program will be initiated and maintained on a 5-year cycle.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Demin Water	Operating Time	30 Days Note 1	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4897	Temperature (°F)	50-90	50-90	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: 830060R	Relative Humidity (%)	80	80	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Demin Water to Containment	Radiation (Rad)	7.6×10^4	1.0×10^6	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Turbine Building Pipe Tunnel; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation N/A; Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
	1. Not required to operate for HELB outside containment.

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Sheet 2

Component ASCO Solenoid Valve SV-4897

This solenoid valve isolates demineralizer water to the containment building via manual operation from the control room. It is backed up by an isolation check valve which is located inside containment. The control valve closes when the solenoid valve is de-energized. Based on discussions with ASCO, in all probability when in the de-energized position, failure of the solenoid valve elastomers will result in the valve remaining in the fail-safe mode. Since the valve must be manually operated to close following a LOCA, the emergency procedure will have to be modified to identify this action to the operator. This manual operation will be accomplished within one hour after the event. After closure the valve will remain in its de-energized position for 30 days.

Temperature, pressure, and relative humidity all remain at ambient conditions outside containment. The total integrated 30-day radiation exposure at the containment surface is 7.6×10^4 rads which is less than the threshold limit of greater than 1×10^6 rads for the valve elastomer BUNA N.

The aging of the elastomers is not significantly accelerated during the post-LOCA period. Replacement of aged and/or worn components on a periodic basis will be done to maintain a qualified life for the valve. The program will begin by June 30, 1982.

Based on the normal environment it encounters with respect to temperature, pressure and humidity, the ability to withstand accident radiation, and because it is only necessary to de-energize the valve following a LOCA, the solenoid can be assumed qualified. Aging affects the nonmetallic components but they will be replaced. The backup check valve is leak tested each refueling outage and will maintain the containment boundary, and it also provides an acceptable reason to maintain operation. With the preventative maintenance replacement program the valve will be qualified.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Main Steam	Operating Time	30 Days (Note 1)	Sheet 2	Section II.D	See Sheet 2	Evaluation	
Plant I.D. Number: SV-4899	Temperature (°F)	50-100	50-100	Section II.E	See Sheet 2	Evaluation	
Component: Solenoid Valve	Pressure (PSIA)	14.7	14.7	Section II.E	See Sheet 2	Evaluation	
Manufacturer: ASCO; Model Number: 830060R	Relative Humidity (%)	80	80	Section II.E	See Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Steam-to-Steam Jet Air Ejectors	Radiation (Rad)	1.6×10^5	7.0×10^6	Section II.E	1	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Evaluation	
Location: Turbine Pipe Tunnel; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation; Above Flood Level: Yes No							

DOCUMENTATION REFERENCES	NOTES
1. The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	1. Not required to operate for HELB outside containment.

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System: Main Steam	Operating Time	30 Days (Note 1)	See Sheet 2	Section II.D	See Sheet 2	Evaluation	
Plant I.D. Number: SV-4916	Temperature (°F)	50-100	50-100	Section II.E	See Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	14.7	14.7	Section II.E	See Sheet 2	Evaluation	
Model Number: 830060R	Relative Humidity (%)	80	80	Section II.E	See Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Turbine Bypass Warm-Up Line Valve	Radiation (Rad)	1.6×10^5	7.0×10^6	Sections II.C and II.E	1	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Evaluation	
Location: Turbine Building Pipe Tunnel; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation; Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	1. Not required to operate for HELB outside containment.

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Sheet 2

Component ASCO Solenoid Valves SV-4899-4916

Both of these solenoids pilot control valves which are backup valves to the main steam isolation valve. They are not required to operate in an HELB outside containment. The valves will be required to be manually operated as they do not receive a safety system initiation. There also has never been any credit taken for the operation of these valves in any safety analysis. Reliance has been placed solely on the MSIV. Immediate operator action after he has determined the symptoms of a break are to close the automatic isolation valves via a hand switch. This will either back up or initiate their closure. Since these valves are not automatically actuated, the procedure will have to be modified such that their common hand switch is also actuated by the operator. Operator action is assumed as 10 minutes. The valves must, therefore, remain operable for this period of time. For the 30-day period, the valves will remain de-energized with the control valves shut.

The normal and post-LOCA environments with respect to pressure and humidity are at normal nonharsh conditions. Higher than normal temperatures in the area near the solenoids are, when the Plant is at power, the result of the nearby main steam line. These temperatures will remain high through the time the valves are required to close but will decay to ambient shortly thereafter. Both valves are normally energized when the Plant is at power. When de-energized, the solenoid vents the air from the control valve, closing it. In its de-energized position with the exhaust port open, the solenoid is in the fail-safe mode, according to ASCO. A failure of the BUNA N seats will not result in actuation of the control valve.

The radiation dose accumulated for a 40-year life is approximately 1.4×10^5 rads. In the 10 minutes until the time to operate the valve, there will be no significant additional exposure. The 30-day, post-LOCA integrated dose is 2.0×10^4 rads which results in a total lifetime exposure of 1.6×10^5 rads. This falls within the radiation 25% damage limit of 7.6×10^6 rads, based on compression set, for the BUNA N elastomers used in the valve.

BUNA N is known to be susceptible to the effects of thermal aging; it shows a good resistance to radiation. As discussed earlier, the temperatures that the valves are exposed to during power operation are above normal. Presently, the valves are not on a preventative maintenance part replacement program to replace the age susceptible parts. This will be done, however, by June 30, 1982, and be continued on a five-year program. The failure mode of the valves due to deterioration of the BUNA N seats would be leakage through the seats resulting again in a fail-safe operation of the control valve.

Based on the fail-safe failure mode of the valve and because no credit in the safety analysis has been taken for these valves, they remain acceptable as is. However, to maintain correct operability of the valve, it will be maintained on a preventative maintenance schedule initiated by June 30, 1982.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Depressurization	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-4980, SV-4981 SV-4982, SV-4983	Temperature (°F)	235	Sheet 2	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Manufacturer: ASCO	Relative Humidity (%)	100	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: HTX831677	Spray	Lake Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	Sheet 2	Evaluation	
Function/Service: Reactor Depress Isolation Valve	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Submergence	Not Subject To Submergence	-	-	-	-	
Location: Containment							
Elevation: 667							
Flood Level Elevation 590							
Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES

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Sheet 2

Component ASCO Solenoid Valves SV-4980 Through 4983

These solenoid valves are pilot valves for the reactor depressurization system (RDS) isolation valves. The valves are actuated on RDS initiation signals from low steam drum level, fire pump start (core spray water), low reactor water level and a two-minute time delay. The solenoids are normally de-energized and are energized on RDS initiation. The RDS function is for small to intermediate size breaks to relieve reactor pressure so that the low-pressure core spray system can provide core cooling. Once open, the isolation valves shall remain open; however, for most all breaks requiring RDS, the reactor becomes depressurized and there will be no increase of pressure in the primary system above the core spray injection pressure; ie, pressure will be relieved out the break and, therefore, the RDS would not be required except for the initial depressurization. Only for a very small break would it be necessary for the valves to remain operable and only one of the four valve trains would then be required.

The valves have been recently modified to replace the Celcon plastic disc holder with a metal disc holder. The solenoid is equipped with a Class H high-temperature coil. Class H coils are also used in ASCO's NP series solenoids that have passed type testing to meet the IEEE-323 (1974) requirements. ASCO Test Report AQS21678/TR, Rev A. The coils are suitable for ambient temperatures up to 212°F for continuous operation. Temperature is above 212°F for only about 10^3 seconds for the spectrum of breaks shown in Figure 1 of Section II. The coil is also radiation resistant (the ASCO test irradiated the coils to 2×10^8 rads). The coil is also suitable in high humidity conditions. The containment sprays are not likely to impinge on the valves as they are located above and behind the nozzles. The watertight enclosures are suitable to keep both humidity and water spray, which is not caustic, out of the coil area. The explosionproof enclosure will also withstand the LOCA pressure of 41.7 psia.

The BUNA N valve elastomers have a radiation resistance of 7.0×10^6 based on 25% degradation due to compression set. This is a factor of 10 larger than its lifetime plus LOCA dose. The coils, as stated above, have been tested to 2×10^8 rads.

The valves were installed in the Plant in 1976 and by June 30, 1982 will have not aged significantly as they are normally de-energized except for testing and are in a nonharsh, normal environment. Due to lack of qualification type testing, they will be replaced by the June 30, 1982 date. Presently, they are acceptable because of their materials construction to withstand the six-year life plus a 30-day post-LOCA period.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Depressurization Plant I.D. Number: SV-4984, -4985, -4986, -4987 Component: Manufacturer: Target Rock Model Number: 73V001 (6"x10") Purchase Order Number: Function/Service: Reactor Depressurization Valve Accuracy: Spec: Demo: Location: Containment Elevation: 665 Flood Level Elevation 590 Above Flood Level: Yes X No:	Operating Time	30 Days	Sheet 2	Section II.D	Sheet 2	Test and Evaluation	
	Temperature (°F)	235	300	Section II.E	1	Test	
	Pressure (PSIA)	41.7	85	Section II.E	1	Test	
	Relative Humidity (%)	100	100	Section II.E	1	Test	
	Spray	Not Subject To Spray	-	-	-	-	
	Radiation (Rad)	7.3×10^5	3.3×10^7	Section II.E	1	Test	
	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Test and Evaluation	
	Submergence	Not Subject to Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. Environmental Test Report on TRC Model 72V, Type 1" "Y" Pattern Solenoid Motor Valve Report No 1500, and EAST-WEST Technology Corp Report No 798-4 of TRC Model 73E-001	

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Sheet 2

Component Target Rock Solenoid Operated Valve SV-4984 Through SV-4987

These solenoids pilot the reactor depressurization system (RDS) depressurizing valves. The valves were installed in 1976. The valves are actuated by RDS initiation signals from low steam drum level, fire pump start (core spray water), low reactor water level and a two-minute time delay. On RDS initiation, the valves are energized. The function of RDS is to depressurize the primary system so that the low-pressure core spray system can provide core cooling. Once energized, the valves will remain energized throughout the post-LOCA period; however, for most all breaks requiring RDS, the reactor becomes depressurized and there will be no increase of pressure in the primary system above the core spray injection pressure; ie, pressure will be relieved out of the break and, therefore, the RDS would not be required except for the initial depressurization. Only for a very small break would it be necessary for the valves to remain operable and only one of the four valve trains would then be required.

The valves are mounted above and behind the containment spray nozzle distribution pattern and are, therefore, not subject to spray. Big Rock Point's containment sprays are, at any rate, water from Lake Michigan and are not corrosive and would not be expected to be detrimental to the valve operation. Furthermore, the solenoids have NEMA Type 4 waterproof enclosures.

Reference 1 describes testing of target rock solenoid operated valves. The test specimens were production model valves of the similar design using the same materials and solenoid coil type as was used in the Big Rock Point model valves. The testing also included the NEMA Type 4 junction box with Kulka terminal board which is the same as on the installed valves. The unit was preirradiated to 33 megarads, aging was conducted as a temperature-humidity wear-out test and the valve was heated by passing fluid through it to 200 ± 10°F with the test chamber air temperature at 150 ± 10°F and relative humidity of 55 ± 5%. The valve was then cycled 1000 times. The chamber conditions were maintained for 240 hours. The accident simulation test temperature and pressure increased to 300°F and 70 psig in 10 seconds then to 340°F within 5 minutes where it was held for 3 hours and then reduced to ambient in 2 hours. This was then repeated. Relative humidity was maintained at 100% and demineralized water spray was used. After the repeat, the test chamber was maintained at 250°F/25 psig for 4 hours. The pressure was then reduced to ambient and temperature maintained at 200°F for 30 days.

Although the preaging did not simulate a 40-year life, the LOCA simulation temperatures being significantly above the Big Rock Point break envelope have enough margin to provide additional aging data of the test assembly. Big Rock Point's maximum temperature of 235°F lasts for only a few seconds, temperatures above 200°F for the large break (Figure 1) are only for a 1000 second period and, lastly, atmospheric temperatures are reached within one day. 200°F for 30 days by the 10°C rule provides about 3-years' simulated life at 104°F.

Sheet 2 (Contd)

The valve elastomers are ethylene-propylene rubber, silicone rubber and an asbestos and rubber insulator washer. The asbestos and rubber washer is captured above and below the solenoid coil. Although it may deteriorate in its life, it will remain intact due to its position. The EPR and silicone rubber O-rings are also captured and their failure will not affect the operation of the valve.

Based on the testing of the target rock solenoid valve with the same NEMA Type 4 enclosure, Class H coil and nonmetallic materials, the valves are considered acceptable for their remaining lifetime. Three of the four pilot solenoids are scheduled for replacement in the upcoming refueling outage.

The aging of the materials will not be detrimental to the operation of the solenoid and, at normal ambient temperatures with the solenoid normally de-energized, the materials are not subjected to an accelerated aging process.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION

System: Containment Ventilation

Plant I.D. Number: SV-9151, 9152

Component: Solenoid Valve

Manufacturer: ASCO

Model Number: 8316C44

Purchase Order Number:

Function/Service: Containment Supply Ventilation Isolation Valves

Accuracy: Spec:
Demo:

Location: Sphere Ventilation Room

Elevation:

Flood Level Elevation: N/A

Above Flood Level:
Yes: No:

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	40-90	40-90	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	20-100	20-100	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject To Spray	-	-	-	-	
Radiation (Rad)	1.9×10^5	7.0×10^6	Section II.E	1	Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Not Subject To Submergence	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES
1. The Effects of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	Parameter	Environment: Accident	Environment: Qualification	Documentation References: Accident	Documentation References: Qual	QUALIFICATION METHOD	OUTSTANDING ITEMS
System: Containment Ventilation	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-9153, 9154	Temperature (°F)	40-90	40-90	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: FT-8316D44	Relative Humidity (%)	20-100	20-100	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Containment Exhaust Ventilation Isolation Valves	Radiation (Rad)	1.9×10^5	7.0×10^6	Section II.E	1	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Sphere Ventilation Room; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	

Flood Level Elevation: N/A
Above Flood Level: Yes: No:

DOCUMENTATION REFERENCES	NOTES
1. The Effects of Nuclear Radiation on Flastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	

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Sheet 2

Component ASCO Solenoid Valves SV-9151 Thru 9154

These valves are the pilot valves for the containment ventilation supply and exhaust valves. They close on all scrams and containment high radiation. They may be required to reopen in the event of a containment vacuum condition. The valves are located outside containment and are subjected to normal ambient temperature, pressure and relative humidity following a LOCA. Radiation during a 30-day exposure will be 1.9×10^5 rads which is less than the 7.0×10^6 rads for a 25% damage valve of the BUNA N elastomers. The valves presently undergo a preventative maintenance program to replace the valve elastomers. A change out is scheduled for the upcoming 1980 refueling outage. Based on the valves being located in a normal environment, both before and after a LOCA, and the present PM schedule, the valves are considered qualified for a 40-year plus LOCA lifetime.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Containment Ventilation	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SV-9155, 9156	Temperature (°F)	90	90	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	14.7	14.7	Section II.E	Sheet 2	Evaluation	
Model Number: WP-80306	Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Ventilation Probe	Radiation (Rad)	1.9×10^5	7.0×10^6	Section II.E	1	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Sphere Ventilation Room; Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation: N/A; Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. The Effects of Nuclear Radiation on Elastomeric and Plastic Components and Materials, R W King, et al, REIC Report No 21.	

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Sheet 2

Component ASCO Solenoid Valves SV-9155, 9156

These solenoid valves are the ventilation probe valves. They close on all scrams. The valves are in series and are located outside containment. They will de-energize and close, in their fail safe position, on a scram signal well within one hour from a break and remain closed throughout the post-LOCA period. The valves are direct acting solenoid-operated valves which provide containment boundary when closed. Containment pressure aids the seating of the valves. Following a break the valves will remain in normal atmosphere with respect to temperature pressure and relative humidity. Radiation during the 30-day post-LOCA period will be 1.9×10^5 rads which is less than the 7.0×10^6 rads determined to be the 25% damage valve for the BUNA N seating material. These valves will be put into the preventative maintenance program to change out the valve elastomers. Local leak rate testing of these valves has shown no apparent deterioration of the valve seats. Because the valves operate in a normal environment, will not be subject to damaging radiation and presently provide an adequate containment boundary, the valves are acceptable as is. The preventative maintenance program will be initiated by June 30, 1982 to replace the age susceptible materials.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	Parameter	Environment: Accident	Environment: Qualification	Documentation References: Accident	Documentation References: Qual	QUALIFICATION METHOD	OUTSTANDING ITEMS
System: Reactor Protection	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SVNC-22F, G, H, J	Temperature (°F)	235	235	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve; Manufacturer: ASCO	Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Model Number: 831622 (Note 1)	Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Purchase Order Number:	Spray	Not Subject to Spray	-	-	-	-	
Function/Service: Dump Tank Vent Valves	Radiation (Rad)	4.2×10^6	4.2×10^6	Section II.E	Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	5 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Location: Containment; Elevation: 580'-0"	Submergence	Yes	Sheet 2	Section II.E	Sheet 2	Evaluation	
Flood Level Elevation: 590'-0"; Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
	1. Valve contains HVA-90-441-1A pilot head assembly and HVA-90-405-2A spare parts kit.

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Sheet 2

Component ASCO Solenoid Valves SVNC-22F, G, H, J

The scram dump tank vent valves prevent pressure buildup in the scram dump tank which may, if the control rod drive is isolated, cause withdrawal of a control rod when the dump tank pressure would exceed reactor pressure. The valves are normally energized and, on a scram signal, they de-energize after a one-minute time delay. Once de-energized, the valves remain in this position. A scram following a break would be initialed either automatically on low drum level or high containment pressure or, in the event of a small break, operator action would initiate a scram. Time for operator action is 10 minutes. The solenoids would have to, because of the time delay, operate in the 235°F temperature, 41.7 psia pressure and 100% relative humidity.

The valve being normally energized has internal core temperatures in excess of 200°F. The BUNA N and Zytel materials are subjected to these internal temperatures. The LOCA environment will not subject the valve materials to higher than normal temperatures during the minute it must remain energized. The LOCA pressure should also not affect the operation of the valve. ASCO solenoids are routinely pressurized in containment leak rate testing without affecting the operation of the valves. Relative humidity of 100% will also not affect the watertight NEMA type of enclosure.

The valve, once de-energized, must remain in this state to maintain the dump tank vent path. The valves must then fail-safe and the effects of radiation aging and submergence should not affect them. An integrated gamma dose for two hours in air of 2.0×10^5 rads plus the beta-gamma dose in water for 30 days of 4.0×10^6 rads yields 4.2×10^6 rads which is within the damage limit of [illegible]ceptible materials. ASCO states B[illegible] N is acceptable to 7.0×10^6 ra[illegible]ytel 103 to 5.0×10^6 rads. NRC G[illegible]ines, Table C-1, state [illegible]eoprene acceptable to 10^7 rads. The most [illegible]gnificant aging of the valves is not in the post-LOCA atmosphere, but rather during the normal energization of the valve. This, in fact, led to a preventative maintenance replacement program to replace the materials on a five-year schedule. The high post-LOCA temperatures, as stated above, will not subject the materials to additional stress during the time it takes for them to operate. ASCO has also stated BUNA N is acceptable in continuous temperatures up to 180°F and 240°F for short periods. The materials in the post-LOCA atmosphere will not be exposed to temperatures in excess of these. Submergence also should not cause a failure mode which will cause opening of the control valve. With the solenoid de-energized, failure of the diaphragms and seals will only result in fail-safe operation according to ASCO.

In summary, the valves have been evaluated to be acceptable for post-LOCA operation. They, however, do not meet the guideline requirements and, because of this and the need to maintain the venting of the dump tank, they will be replaced by June 30, 1982.

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

EQUIPMENT QUALIFICATION REPORT

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION
System: Reactor Protection
Plant I.D. Number: SVNC-27 (Note 1)
Component: Solenoid Valve
Manufacturer: ASCO
Model Number: 831622 (Note 2)
Purchase Order Number:
Function/Service: Scram Pilot Valves
Accuracy: Spec: Demo:
Location: Containment
Elevation: 586'-0"
Flood Level Elevation: 590'-0" **Above Flood Level:** Yes: No: X

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	10 Minutes	10 Minutes	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	235	235	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject to Spray	-	-	-	-	
Radiation (Rad)	2.0×10^5	2.0×10^5	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Yes	Sheet 2	Section II.E	Sheet 2	Evaluation	

DOCUMENTATION REFERENCES	NOTES
1. CP Co response to IE Bulletin 78-14, "Deterioration of BUNA N Components in ASCO Solenoids," February 2, 1979.	1. SVNC-27A2A thru SNVC-27F5B; 64 total. 2. Valve contains HVA-90-441-1A pilot head assembly and HVA-90-405-2A spare parts kit.

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Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

EQUIPMENT QUALIFICATION REPORT

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION

System:
Reactor Protection

Plant I.D. Number:
SVNC-22A, B

Component:
Solenoid Valve

Manufacturer:
ASCO

Model Number:
831622 (Note 1)

Purchase Order Number:

Function/Service:
Master Scram Valves

Accuracy: Spec:
Demo:

Location:
Containment

Elevation:
578'-0"

Flood Level
Elevation 590'-0"
Above Flood Level:
Yes: No: X

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	10 Minutes	10 Minutes	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	235	235	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Spray	Not Subject to Spray	-	-	-	-	
Radiation (Rad)	2.0×10^5	2.0×10^5	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Submergence	Yes	Sheet 2	Section II.E	Sheet 2	Evaluation	

DOCUMENTATION REFERENCES

1. CP Co response to IE Bulletin 78-14, "Deterioration of BUNA N Components in ASCO Solenoids," February 2, 1979.

NOTES

1. Valve contains HVA-90-441-1A pilot head assembly and HVA-90-405-2A spare parts kit.

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Protection	Operating Time	10 Minutes	10 Minutes	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: SVNC-22C, D	Temperature (°F)	235	235	Section II.E	Sheet 2	Evaluation	
Component: Solenoid Valve	Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Manufacturer: ASCO	Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Model Number: 831622 (Note 1)	Spray	Not Subject to Spray	-	-	-	-	
Purchase Order Number:	Radiation (Rad)	2.0×10^5	2.0×10^5	Section II.E	Sheet 2	Evaluation	
Function/Service: Dump Tank Isolation Valves	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Submergence	Yes	Sheet 2	Section II.E	Sheet 2	Evaluation	
Location: Containment							
Elevation: 578'-0"							
Flood Level Elevation 590'-0" Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. CP Co response to IE Bulletin 78-14, "Deterioration of BUNA N Components in ASCO Solenoids," February 2, 1979.	1. Valve contains HVA-90-441-1A pilot head assembly and HVA-90-405-2A spare parts kit.

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Sheet 2

Component ASCO Solenoid Valves SVNC-22A, B, C, D and SVNC-27A2A Thru F5B

These solenoid valves, the master scram, the scram valve solenoids and scram dump tank isolation valve solenoids operate automatically on a reactor protection signal or by manual action. In the event of a small break LOCA where the operator observes symptoms of a LOCA and takes manual action to scram the reactor, the operation time is considered to be ten minutes. In the event of a large break LOCA, the scram will be automatic and within seconds after the break. Following the scram, it is inconsequential if the valves fail in the LOCA environment.

These valves have had replacement spare parts kits installed in 1979. A thermal aging problem has been determined to exist due to the solenoids being constantly energized resulting in the components being subjected to a constant heat source. The elastomers, BUNA N, neoprene and Zytel 103, are the materials replaced in the five-year preventative maintenance program. Being constantly energized, the solenoid enclosure skin temperature has been measured to be approximately 125°F. Internal temperatures within the core range above 200°F.

Based on the data provided by General Electric in Reference 1, this model valve can operate in sustained periods of up to seven years without failure in the energized mode.

For automatically initiated scrams, these solenoids will be actuated by the safety system before the temperature or pressure has risen much above ambient; eg, at three seconds (Figures 1 and 2) for the large break, the temperature is 150°F and the pressure is 21 psia. The internal valve materials will not be affected by the LOCA temperature. If operator action is required to initiate a scram in event of a LOCA, the break would be small and both temperature and pressure would not have risen much above ambient. The result is that these valves will be exposed to a less severe environment than noted on the qualification sheet. Furthermore, ASCO has stated the valve materials can withstand the peak LOCA temperatures. These are not, of course, significantly larger than the internal valve temperatures. ASCO has also stated the solenoid enclosures are of sufficient strength to withstand the LOCA pressure of 41.7 psia without affecting the operation of the valve. The solenoid has a NEMA 4 watertight enclosure and, therefore, 100% relative humidity will also not be a factor in the operation of the valve.

A two-hour gamma radiation dose of 2.0×10^5 rads in air is conservatively used for the integrated 10-minute dose. The actual dose would, of course, be much less and, in fact, be negligible since, in any event, scram will occur long before the core is uncovered. The valve nonmetallic materials all have a threshold damage limit greater than the two-hour integrated dose. Radiation will, therefore, not affect the valve operation.

Aging effects on the valve will not be significantly accelerated in the LOCA event, but do occur as stated above due to the continuous energization of the solenoid. The valves have been placed on a five-year program to replace the

Sheet 2 (Contd)

nonmetallic components. The five-year program was established in response to IE Bulletin 78-14. Based on the replacement program, the valves are considered qualified for a 40-year plus LOCA lifetime.

The valves become submerged at some time after the accident. Submergence, however, will not occur prior to a scram. After a LOCA and scram have occurred, the position of the control valves becomes inconsequential; therefore, submergence also becomes unimportant.

In summary, the valves are considered qualifed based on their 10-minute operating time and component replacement program which will maintain the valve qualification. Even though they do not meet guideline requirements for equipment inside containment, they are acceptable as is.

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

EQUIPMENT QUALIFICATION REPORT

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION
System: Post-Incident
Plant I.D. Number: As Listed in Equip List Section II.C
Component: Junction Boxes
Manufacturer:
Model Number:
Purchase Order Number:
Function/Service: Wire Termination
Accuracy: Spec: Demo:
Location: Containment
Elevation: >591
Flood Level Elevation Above Flood Level: Yes: No

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	1 Hour	30 Days	Section II.D	Sheet 2	Evaluation	
Temperature (°F)	235	235	Section II.E	Sheet 2	Evaluation	
Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Relative Humidity (%)	100	100	Section II.E	Sheet 2	Evaluation	
Spray	Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Radiation (Rad)	7.3×10^5	7.3×10^5	Section II.E	Sheet 2	Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Sheet 2	Evaluation	
Submergence	None	-	-	-	-	

DOCUMENTATION REFERENCES	NOTES

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Sheet 2

Component Junction Boxes

In the LOCA environment, the junction boxes provide protection from containment water spray in addition to their normal function of keeping dirt and dust out of the terminal connection. In 1975, Consumers Power conducted an inspection of all junction boxes inside containment. At that time, it was determined some may not withstand the pressure attained during a LOCA. To equalize the pressure differential, all boxes were provided with, or judged to have, an adequate vent to the containment atmosphere. Where boxes were determined to have the possibility to fill with condensate, a drain hole was provided in the bottom of the box. These noles may serve the dual purposes of both providing a vent and/or drain opening. The junction boxes may or may not be equipped with watertight covers. However, it is not necessary to be watertight because of the existence of the drain holes.

Terminal blocks, splice connections and other types of terminal connections may be found in these junction boxes. These connections are discussed elsewhere in this report.

With the existence of drain and/or vent holes, the junction boxes are acceptable.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK OINT
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Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: To Safety-Related Equipment	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: –	Temperature (°F)	235	235	Section II.E	1	Test and Evaluation	
Component: Splice Insulation/ Waterproofing	Pressure (PSIA)	41.7	41.7	Section II.E	1	Test	
Manufacturer: 3M	Relative Humidity (%)	100	100	Section II.E	1	Test	
Model Number:	Spray	Water	Water	Section II.E	1	Test	
Purchase Order Number:	Radiation (Rad)	7.3×10^5	$>7.3 \times 10^5$	Section II.E	2, Sheet 2	Evaluation	
Function/Service: Cable Splice Taping	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Evaluation	
Accuracy: Spec: Demo:	Submergence	None	–	–	–	–	
Location: Inside/Outside Containment							
Elevation:							
Flood Level Elevation 590							
Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Preliminary Test Report - Cable Splice Waterproof Environmental Testing Nuclear Services Corporation, September 8, 1978.	

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: Big Rock Point
Docket: 50-155

Component Sheet No.:
Revision:
Date:

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Letter J R More, 3M Company, to W Hall Consumers Power Company, October 20, 1978, Radiation Test Results.	

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Sheet 2

Component Splices

The splices connecting the penetration pigtails to the cables were covered with waterproofing tape as part of the fire protection modification performed in 1978. The splice now only provides the mechanical and electrical connection between the two cables. The waterproofing tape provides the qualification.

The tape used in the test and in the plant was Scotch Brand 23 (ethylene-propylene material) wrapped first with Scotch Brand 33+ (a vinyl material) wrapped on top. To assure proper adhesion, the cables were first prepared using the kit specified by 3M. The preparation and taping was done in accordance with strict procedure.

The waterproofing tape was environmentally tested. The test specimens were spliced conductors whose splices were covered with waterproof tape and were placed in the test chamber. A dielectric test was performed on each spliced cable and then the cable was loaded to 6 amps. Steam was introduced into the test chamber and the conditions of 235°F and 41.7 psia were reached in 10 seconds. The temperature was held for 1 hour. The conditions were allowed to decay (held to a linear decay) to 115°F and 16.9 psia in a 24-hour period. These conditions were held for four days. While the chamber was sealed, a dielectric test was repeated. The specimens were removed from the chamber and sprayed for 20 minutes with water from a 1-1/2" fire hose with a head pressure of 80 psig. The dielectric tests were then repeated. The spliced cables passed all of the tests.

The radiation resistance of the tape is as follows: at 5×10^7 R, dielectric was essentially the same and elongation at break was approximately 1/2 of the values at zero radiation.

No age related data could be obtained. The splices and tape are in an environment of 50°F to 90°F (seasonal variation).

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The splices, which are covered with waterproof tape, are considered qualified because of the following: The tape was environmentally tested to a temperature-pressure envelope that matched the required envelope for the first hour and exceeded the required envelope for the next 19 hours. The test duration was not as long as required; however, the test exceeded the envelope through most of the test and at the time of which the test was terminated, the containment would have been less than 125°F. The spray test was much more severe and the degradation due to radiation was acceptable.

The effects of thermal aging on these materials will be investigated by June 30, 1982; however, since the materials are relatively new (fall 1978 installation) and since the normal environment averages 70°F, these splices are concluded to be acceptably qualified in the interim.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Depressurizing	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number:	Temperature (°F)	235	340	Section II.E	1	Test	
Component: Terminal Blocks	Pressure (PSIA)	41.7	117.7	Section II.E	1	Test	
Manufacturer: General Electric/ States Company; Model Number: CR-151/NT	Relative Humidity (%)	100	100	Section II.E	1	Test	
Purchase Order Number:	Spray	Water	Sheet 2	Section II.E	See Sheet 2	Evaluation	
Function/Service: Wire Termination	Radiation (Rad)	7.3×10^5	Sheet 2	Section II.E	2	Evaluation	
Accuracy: Spec: Demo: Location:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Section IV & 1	Test and Evaluation	
Containment; Elevation: >590	Submergence	None	-	-	-	-	
Flood Level Elevation: 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Letter M Costandi, GE, to J Yope, CP Co, dated February 21, 1979, G-EJ-9-17, Electrical Terminal Block Testing.	1. GE Type CR-151 and States Company, Type NT.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. "Effects of Radiation on Materials and Components," Kircher and Bowman.	

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Sheet 2

Component Terminal Blocks

The GE Type CR-151 and States Company Type NT terminal blocks were tested as part of GE's electrical penetration qualification test program.

The terminal blocks were mounted in a test chamber. The insulation resistance was measured using 500 V dc at ambient conditions. The blocks were then subjected to a LOCA environment during which the insulation resistance was measured once per day. The test sequence was as follows: The environment was held at 260°F and 21 psig for 1.5 days (preaging); then the pressure and temperature were increased to 320°F and 75 psig and held for 1.5 hours; increased to 340°F and 103 psig and held for 3 hours; decreased to 320°F and 75 psig and held for 4.5 hours and then decreased to 260°F and 21 psig and held for 8 days. The preaging simulates 40-year life for most materials used in terminal blocks. The temperature and pressure greatly exceed that needed to demonstrate qualification and the test duration is longer than required. Reference 1 also stated that the insulation resistance was lowered but remained at a sufficient level to assure continued function of the electrical equipment without circuit overload.

Spray was not used during the test. These terminal blocks are located in sealed junction boxes and, therefore, are not affected by the sprays (see junction box write-ups). The radiation at the center of containment is 7.3×10^5 rads over 30 days. The terminal blocks are in sealed junction boxes. The Big Rock Point con ment is split into many rooms separated by concrete walls. The terminal cks are mounted on these walls. Also, ambient conditions are reach in 3 days, not 30. For these reasons, the total integrated dose these blocks will see is much less than 7.3×10^5. The terminal blocks will be able to withstand these relatively low doses as most of the materials of construction can withstand 10^6 rads.

The terminal blocks were tested in an environment more severe and of longer duration than that required to demonstrate qualification. The blocks were preaged prior to the test and are evaluated to be able to withstand the radiation. Therefore, they are qualified.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Note 1	Operating Time	30 Days	30 Days	Note 1	Sheet 2	Evaluation	
Plant I.D. Number:	Temperature (°F)	235	340	Section II.E	2	Test	
Component: Terminal Blocks	Pressure (PSIA)	41.7	120.7	Section II.E	2	Test	
Manufacturer: Westinghouse	Relative Humidity (%)	100	100	Section II.E	2	Test	
Model Number: 542247, 805432	Spray	Water	Borated Water	Section II.E	2	Test	
Purchase Order Number:	Radiation (Rad)	7.3×10^5	1×10^7	Section II.E	3	Evaluation	
Function/Service: Wire Termination	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Section IV & 3	Evaluation	
Accuracy: Spec: Demo:	Submergence	None					
Location: Containment							
Elevation: >590							
Flood Level Elevation 590							
Above Flood Level Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Westinghouse Descriptive Bulletin 34-350, Page 5. 2. Test Report on the Effect of a LOCA on the Electrical Performance of Four Terminal Blocks PEN-TR-77-83, September 13, 1977, Westinghouse.	1. These terminal blocks have been purchased but not installed. Environmental Qualification data is presented in the event that these are installed at some later date.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Westinghouse Terminal Block Material Evaluation, A P Colaiaco. 4. Letter W Skibitsky to J Keppler I&E, February 13, 1978. Response to IE Bulletin 78-02.	

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Sheet 2

Component Terminal Blocks

The terminal blocks are cellulose filled phenolic made by Westinghouse. Model 542247 is the same as Model 805432 but with a black marking strip per Reference 1. At Big Rock Point, all terminal blocks are covered by splash-proof metal covers.

These blocks were environmentally tested. The blocks were mounted to a fixture which was attached to a header through which copper feed throughs were passed. The test consisted of applying 600 V ac to the terminals and filling the test chamber with steam. Borated water was sprayed at the rate of 0.32 gpm for 1 hour. The environmental conditions were as follows: 340°F and 106 psig for 2 hours; decreasing to 329°F and 93 psig at 3 hours (borated water introduced at this time); rising slightly to 332°F and 91 psig at 4 hours; maintained until 5.5 hours; allowed to cool (at 21.5 hours the temperature was still above ambient) and opened at 26.5 hours. The insulation resistance decreased but the blocks were able to function at 600 V ac. The test duration does not cover the time until ambient is reached at Big Rock Point but is still considered to be an applicable test because the temperatures reached were much higher than that at Big Rock Point and for a much longer period of time. Also, the test report pointed out that at 21.5 hours, the temperature was still above ambient. At this time, during the Big Rock Point LOCA, the temperature pressure envelope shows the temperature at 130°F; however, the actual analytical break curves would show temperature below that. Neither of these temperatures are significantly above ambient so that the test very closely matched the end of the temperature profile. Lastly, the materials of construction are not age sensitive. Reference 3 shows that the qualified life is 40 years at 230°F. Therefore, a shorter test duration is acceptable.

The materials of construction were evaluated in Reference 3 to be capable of withstanding 2×10^7 rads. This is much higher than the gamma dose received in an accident. Beta radiation will not affect the terminal blocks as they are housed in splashproof housings. The fact that the spray lasted 1 hour shows that these blocks can withstand the spray. The blocks at Big Rock Point are housed in junction boxes and, therefore, not adversely affected by sprays.

The terminal blocks passed an environmental test with conditions exceeding those found at Big Rock Point for a duration which was judged acceptable. The terminal blocks contain no age sensitive materials and can withstand the radiation. Therefore, these blocks are qualified.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Note 3	Operating Time	30 Days	30 Days	Note 1	Sheet 2	Evaluation	
Plant I.D. Number: -	Temperature (°F)	235	286	Section II.E	1	Test	
Component: Terminal Blocks Manufacturer: General Electric/ Westinghouse	Pressure (PSIA)	41.7	54.7	Section II.E	1	Test	
Model Number: GE-EB-25/805432	Relative Humidity (%)	100	100	Section II.E	1	Test	
Purchase Order Number:	Spray	Water	2640 Ppm Boric Acid	Section II.E	1	Test	
Function/Service: Wire Termination	Radiation (Rad)	7.3×10^5	5×10^6	Section II.E	1	Test	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Section IV & 1	Test	
Location: Containment Elevation: >590	Submergence						
Flood Level Elevation 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Report on Terminal Block/Junction Box Environmental Testing, D B Vail, Northeast Utilities, GEE-78-127.	1. GE Type EB-25. 2. Westinghouse Style 805432.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Letter W Skibitsky to J Keppler, I&E, February 13, 1978. Response to IE Bulletin 78-02	3. These blocks have been purchased but not yet installed. Environmental qualification data is presented in the event that these are installed later.

Sheet 2

Component Terminal Blocks - General Electric, Westinghouse

The terminal blocks were tested by Northeast Utilities. The blocks tested were a GE Type EB-25 made of wood flour filled phenolic and a Westinghouse Style 805432 made of the same material. These blocks were mounted both horizontally and vertically in covered boxes. The terminal blocks at Big Rock Point are also in covered boxes.

The test sequence was as follows: First, the blocks were thermally aged at 150°C for 171 hours. According to the Arrhenius plot, this is equivalent to 40 years at 158°F. Since ambient condition at Big Rock Point is much less than this, the blocks will withstand the aging effects of 40 years plus LOCA. The terminal blocks were then irradiated to 5×10^6 rads. This exceeds the radiation level required to demonstrate qualification.

The environmental test was conducted by using steam to pressurize and heat up the test chamber. The chamber conditions after 8 seconds were 286°F and 40 psig which exceed the accident conditions. These conditions were held for 15 minutes then, to simulate recirculation, the pressure was lowered to 35 psig, held for 60 seconds, then raised to 286°F and 40 psig and held for 31 hours. The conditions were lowered to 232°F and 7 psig over 3 hours and held for 101 hours. These conditions envelope the accident conditions. The blocks passed the test.

Since the blocks were tested to environmental conditions more severe than that at Big Rock Point and were thermally aged prior to the test, it is considered that these are qualified.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Reactor Depressurizing	Operating Time	30 Days	30 Days	Section II.D	Sheet 2	Evaluation	
Plant I.D. Number: Terminal Connections Component:	Temperature (°F)	235	352	Section II.E	1	Test	
Manufacturer: AMP Special Products	Pressure (PSIA)	41.7	135.7	Section II.E	1	Test	
Model Number: PIDG Terminals	Relative Humidity (%)	100	100	Section II.E	1	Test	
Purchase Order Number: Function/Service: Wire Termination	Spray	Water	Boric Acid (3000 Ppm) NaOH (pH of 10.5)	Section II.E	1	Test	
	Radiation (Rad)	7.3×10^5	2×10^8	Section II.E	1	Test	
Accuracy: Spec: Demo: Location:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV & Sheet 2	Evaluation	
Containment Elevation: >590	Submergence	None	-	-	-	-	
Flood Level Elevation: 590 Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Engineering Test Report GPR-575-98, May 14, 1974, AMP, Inc.	

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Component Terminal Connections - RDS

These terminal connections are used in the RDS system and were tested by AMP and Franklin Institute Research Laboratories.

The test specimen is Size 22-16 terminals having polyvinylidene fluoride insulating sleeves. This is the same as was installed at Big Rock Point.

The specimens were exposed to radiation at the rate of 0.5 to 1.0 Mrads/h for an accumulated dose of 2×10^8 rads. The specimens were placed in a test chamber preheated to 130°F for 1/2 hour and then steam was added to the chamber. The temperature rose to 280°F in four seconds and to 350°F in 6 minutes. The pressures corresponded to saturation. A chemical spray solution of boric acid (3,000 ppm) and NaOH (pH of 10.5) was injected 19 seconds after steam started. The sprays used are more caustic than required to demonstrate qualification. The conditions were increased to 352°F and 121 psig, held for 3 hours and then increased to 323°F and 78 psig. These conditions were held for 2 hours, reduced to 253°F and 14 psig and held for 4 days. The chamber was then opened. The terminal connections passed the axial load and dielectric withstand voltage tests conducted after the irradiation and environmental tests. The test conditions enveloped the required environmental conditions both at the peaks and duration. No thermal aging qualification data has been obtained. The connections were installed in 1975 and are in a normal environment of 50-90°F (seasonal variation).

The effects of aging on the materials of construction will be considered by June 30, 1982. Due to the mild ambient temperature and the newness of the materials, these terminal connections are considered acceptable for use in the interim period.

The terminal connections are considered qualified as they passed an environmental test whose conditions were far more severe than those required to demonstrate qualification. The test duration encompassed the time to ambient (3 days for Big Rock Point). The test specimens were similar. The terminal connections are evaluated to have sufficient life.

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EQUIPMENT DESCRIPTION

System: Miscellaneous

Plant I.D. Number:

Component: Note 1

Manufacturer: Anaconda

Model Number:

Purchase Order Number: 32277

Function/Service: Electrical Power & Control Cable

Accuracy: Spec:
Demo:

Location: Containment

Elevation: Some assumed <590

Flood Level Elevation: 590

Above Flood Level: Yes: No: X

ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
Parameter	Accident	Qualification	Accident	Qual		
Operating Time	30 Days	30 Days	Section II.D	1	Test and Evaluation	
Temperature (°F)	235	346	Section II.E	1	Test and Evaluation	
Pressure (PSIA)	41.7	127.7	Section II.E	1	Test and Evaluation	
Relative Humidity (%)	100	100	Section II.E	1	Test and Evaluation	
Spray	Lake Water	Chemical Spray	Section II.E	1	Test and Evaluation	
Radiation (Rad)	7.3×10^5 gamma 1.3×10^7 beta	2×10^8	Section II.E	1	Test and Evaluation	
Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1 & Section IV	Test and Evaluation	
Submergence	Subject To Submergence	Sheet 2	Section II.E	1 & Sheet 2	Test and Evaluation	

DOCUMENTATION REFERENCES	NOTES
1. Letter Dated 12-26-78 From H Kenny of Anaconda to E R Longman of CP Co With Attached Data and FIRL Report #F-C4350-3	1. Electrical Cables With EPR Insulation and Hypolan Jacket - 1/C #1/0, 1/C #6 AWG, 3/C #12 AWG and 1/C #4/0

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Component   Electrical Cables

600 volt class cables, sizes as detailed below, with EPRC (ethylene propylene rubber) insulation and hypolan (chlorosulfonated polyethylene) insulation.

- 1/C #4/0, 19 Strand

- 1/C #1/0, 19 Strand

- 1/C #6 AWG 7 Strand

- 3/C #12 AWG 7 Strand

The above cables are used in Class 1E circuits and are qualified for the severe environments for use inside containment as below:

A. Document Reference 1 demonstrates that the cable is qualified for use inside the containment based on 40-year life and LOCA. The test cycle included the following and meet IEEE-383 (1974) and IEEE-323 (1975) requirements:

   1. Thermal aging at 150°C (302°F) for 168 hours to simulate a 40-year life.

   2. Gamma radiation from a Cobalt-60 source for a total dosage of 2×10^8 R.

   3. LOCA simulation test with chemical spray as below:

      - 346°F at 110 Psig for 8 Hours

      - 335°F at 96 Psig for 3 Hours

      - 315°F at 69 Psig for 4 Hours

      - 265°F at 28 Psig for 81 Hours

      - 212°F at 4 Psig for 26 Days

      The chemical spray has a pH range of 9 to 11 and consisted of 3,000 ppm of boron, 0.064 molar sodium thiosulfate and sodium hydroxide to meet the pH requirements. BRP spray is noncorrosive lake water.

   4. Per Document Reference 1, the cable passed dielectric test when immersed in water. The duration of the test is 26 weeks.

   5. Per FIRL Test Report No F-C4350-3 attached with Document Reference 1, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray

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and this is generally equivalent to submergence. Coincidental high pressure (113 psig) and temperature (346°F) in the LOCA test chamber give rise to a situation which is analogous to submergence under high pressure and temperature.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1, 2, 3, 4, 5	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	235	346	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Component: Note 1 Manufacturer: Cerro	Pressure (PSIA)	41.7	107.7	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Model Number: -	Relative Humidity (%)	100	100	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Purchase Order Number: 34490-1601-20.2	Spray	Lake Water	Chemical Spray	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Function/Service: Electrical Power Cable	Radiation (Rad)	7.3×10^5 Gamma 1.3×10^7 Beta	2×10^8	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1, 2, 3, 4, 5	Test and Evaluation	
Location: Containment Elevation: Some Assumed <590	Submergence	Subject To Submergence	Yes	Section II.E	Sheet 2	Test and Evaluation	
Flood Level Elevation: 590 Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. C of C dated 6/13/75 with Cerro wire test reports. 2. FIRL Test Report No F-C379B, dated March 1974, prepared for Cerro Wire and Cable Co.	1. Electrical cable 1/C #2 AWG with EPR/NEO insulation/jacket materials.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Franklin Institute Research Laboratories (FIRL) Test Report F-C4350-3, dated July 1976, for Anaconda Company. 4. Qualification of Okonite Ethylene Rubber Insulation for Nuclear Plant Service, Okonite Report No NQRN-1. 5. FIRL Test Report F-C5115, dated April 1979, for American Insulated Wire Cable Company.	

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Component Electrical Cable

600 Volt Class Cable 1/C #2 AWG With EPR (Ethylene Propylene Rubber) and NEO (Neoprene) Insulation and Jacketing Materials

The subject cable is used in Class 1E circuits and is qualified for use inside the containment as detailed below:

A. Per Document Reference 1, the cable is qualified for use inside the containment based on 40-year life and LOCA. The test cycle included the following:

1. Thermal aging at 121°C for seven days to simulate 40-year life.

2. Gamma radiation from Cobalt-60 source for a total dosage of 2×10^8 R.

3. LOCA simulation test with chemical spray as below:

   - 346°F at 113 Psig for 6 Hours

   - 335°F at 93 Psig for 3 Hours

   - 315°F at 69 Psig for 4 Hours

   - 265°F at 28 Psig for 81 Hours

   - 212°F at 0 Psig for 26 Days

   The chemical spray has a pH range of 9 to 11 consisting of boric acid (3,000 ppm) and sodium hydroxide to meet the pH requirements. BRP containment spray is Lake Michigan water which is noncorrosive.

4. After LOCA test, the specimens passed the dielectric test when immersed in water at 80 V ac/mil for five minutes.

5. Document References 3, 4 and 5 prove the generic capability of the cables with ethylene propylene rubber (EPR) insulation to meet the severe environments inside the containment.

6. The cable meets the ICEA (formerly IPCEA) standard requirements including long-term moisture absorption test which is done by immersion in water.

7. During LOCA tests described in Paragraph A.3, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (113 psig) and high temperature (346°F) in the LOCA test chamber give rise to a

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situation which is analogous to submergence under high pressure and temperature.

It is concluded that the temperature, pressure, humidity, spray, submergence, radiation and aging requirements have been met and exceeded by substantial margins.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Post-Incident	Operating Time	30 Days	30 Days	Section II.D	1, 2, 3	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	235	325	Section II.E	1, 2, 3	Test and Evaluation	
Component: Note 1 Manufacturer: Kerite Co	Pressure (PSIA)	41.7	96.7	Section II.E	1, 2, 3	Test and Evaluation	
Model Number: Kerite FR	Relative Humidity (%)	100	100	Section II.E	1, 2, 3	Test and Evaluation	
Purchase Order Number: 8093-E-3-AC	Spray	Lake Water	Chemical Spray	Section II.E	1, 2, 3	Test and Evaluation	
Function/Service: Electric Power Cable	Radiation (Rad)	7.3×10^5 gamma 1.3×10^7 beta	1.2×10^8	Section II.E	1, 2, 3	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1, 2, 3	Test and Evaluation	
Location: Containment Elevation: Some Assumed <590	Submergence	Subject To Submergence	Sheet 2	Section II.E	Sheet 2	Test and Evaluation	
Flood Level Elevation: 590 Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. Franklin Test Report #F-2737 2. Letter Dated 3-17-75 From R M Bowman of Kerite Co to J D Westbrook of CP Co	1. Electrical Cables With Kerite FR Insulation and Jacket Material - 1/C #12 AWG, 2/C #14 AWG, 5/C #14 AWG and 7/C #14 AWG

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. Letter Dated 5-29-75 From H E McCrane of Kerite Company to CP Co.	

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Sheet 2

Component Electrical Cables

600 volt class cables, sizes as detailed below, with Kerite FR insulation and jacketing material.

- 1/C #12 AWG

- 2/C #14 AWG

- 3/C #14 AWG

- 7/C #14 AWG

The above cables are used in Class 1E circuits and are qualified for use inside the containment as below:

A. Document Reference 1 details the actual type testing performed under simulated post-accident reactor containment service conditions for the cables with identical insulation and jacket materials as the subject cables. During this test, the cables were subjected to gamma radiation of 1.2×10^8 rads from Cobalt-60 source while being exposed to saturated steam at 82 psig (96.7 psia) and a temperature of 325°F for 13 hours followed by exposure at 5 psig (19.7 psia) at 228°F for 7 days. Borated chemical spray with a pH of 9.5 was sprayed during this test.

   The test specimens were energized during the LOCA test. After the LOCA test, it passed dielectric test (4 kV) when applied for 5 minutes.

   It is concluded that the cable meet or exceed the radiation, temperature, pressure, spray, submergence and humidity requirements as discussed below:

   1. The peak temperature inside the containment of 235°F after an accident at Big Rock exists for less than 1 hour and decreases to 80°F in less than 3 days. The tested temperatures and duration exceed the requirements.

   2. The peak pressure inside the containment of 41.7 psia after an accident exists for less than 1 hour and decreases linearly to 15 psia in less than a day. The tested pressure and duration exceed the requirements.

B. Based on the following evaluation, it is concluded that the cables meet the aging requirements:

   1. The cable meets the physical property requirements of ICEA (formerly IPCEA) Standards such as tensile strength and elongation after air oven test for 168 hours at 121°C.

   2. The radiation aging to a dose of 1.2×10^8 R gamma, far exceeds the requirement of 7.3×10^5 (gamma) and 1.3×10^7 beta. The

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level of beta at the cable insulation surface is decreased by jacket material, conduit and tray cover, etc.

From this higher test value for radiation, it is concluded that the cable will withstand higher temperature for longer duration to qualify for aging requirements.

3. The subject cables were installed in 1970 and the aging requirements (when compared to the rest of the equipment in the Plant) is for less than 32 years.

4. The temperatures and durations during LOCA test detailed in Paragraph A above are much higher than the accident values. This supplements the evaluation given above for aging qualification.

C. Qualification for Submergence, Spray

1. The cable meets the ICEA (formerly IPCEA) Standard requirements, including long-term moisture absorption test which is done by immersion in water.

2. During LOCA tests described in Paragraph A, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (82 psig) and high temperatures (325°F) in the LOCA test chamber give rise to a situation which in analogous to submergence under high pressure and temperature.

Summary

1. Paragraph A provides evidence for temperature, pressure, humidity and radiation qualifications.

2. Paragraph B provides an evaluation for aging qualification.

3. Paragraph C provides an evaluation for spray and submergence.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
Revision:
Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	235	351	Section II.E	1	Test and Evaluation	
Component: Note 1 Manufacturer: Raychem	Pressure (PSIA)	41.7	84.7	Section II.E	1	Test and Evaluation	
Model Number: Flamtrol	Relative Humidity (%)	100	100	Section II.E	1	Test and Evaluation	
Purchase Order Number: 34490-1601-97	Spray	Lake Water	Chemical Spray	Section II.E	1	Test and Evaluation	
Function/Service: Electrical Control Cable	Radiation (Rad)	7.3×10^5 Gamma 1.3×10^7 Beta	2×10^8	Section II.E	1	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1	Test and Evaluation	
Location: Containment Elevation:	Submergence	Subject To Submergence	Sheet 2	Section II.E	1 & Sheet 2	Test and Evaluation	
Flood Level Elevation Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. Franklin Research Institute Test Report No F-C4033-1, dated January 1975.	1. Electrical cables with XLPE insulation and Jacket: 2/C #12 AWG, 2/C #14 AWG, 3/C #14 AWG, 3/C #16 AWG, 6/C #14 AWG, 12/C #16 AWG, 2/C #16 AWG shielded, 2/C #16 AWG twisted, 2/C #10 AWG, 2/C #12 AWG and 3/C #10 AWG.

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Sheet 2

Component Electrical Cables

1. 600 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyclefin) Insulation and Jacketing Material

   - 2/C #14 AWG
   - 3/C #14 AWG
   - 3/C #16 AWG
   - 6/C #14 AWG
   - 12/C #16 AWG
   - 2/C #16 AWG Twisted and Shielded
   - 2/C #16 AWG Twisted

2. 1000 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #12 AWG
   - 2/C #10 AWG
   - 2/C #12 AWG
   - 3/C #10 AWG

The subject cables are used in Class 1E circuits inside the containment in areas not subjected to submergence. The qualification is based on the following documents and discussion:

A. Document Reference 1 demonstrates that the cable with identical insulation is qualified for use inside the containment based on 40-year life and LOCA with chemical spray. In containment, environmental conditions at Big Rock Point are less severe (including the absence of chemical spray) than the parameters to which the cable is type tested. BRP sprays are lake water which is not corrosive. The test cycle included the following:

   1. Thermal preaging at 150°C (302°F) for 25 days and at 160°C (320°F) for 12 days followed by thermal and radiation aging at 150°C (302°F) and 5 x 10^7 R for 7 days.

   2. Combined LOCA and radiation exposure as below:

      - Radiation 1.5 x 10^8 R.

      - Temperature and pressure as follows:

        177°C (351°F) at 84.7 psia for 10 hours; 135°C (275°F) at 45.7 psia for 4.5 days; 100°C (212°F) at 24.7 psia for 26 days.

      The chemical spray during LOCA consisted of 3,000 ppm of boron and 0.064 molar sodium thiosulfate and adjusted with sodium hydroxide to a pH value of between 9.5 to 11.

   3. Dielectric test for five minutes while immersed in water.

Sheet 2 (Contd)

B. Raychem Flamtrol cables meet ICEA (formerly IPCEA) standard requirements including long-term moisture absorption test which is done by immersion in water.

During LOCA tests described in Paragraph A.2, chemical spray was included in the test cycle. The surface of the cable is almost fully covered with fluid for the duration of the spray and this is generally equivalent to submergence. Coincidental high pressure (70 psig) and high temperature (351°F) in the LOCA test chamber give rise to a situation which is analogous to submergence under high pressure and temperature.

It is concluded that the temperature, radiation, humidity, spray, submergence and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for this application.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: BIG ROCK POINT
Docket: 50-155

Component Sheet No:
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Date:

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	100	351	Section II.E	1	Test and Evaluation	
Component: Note 1 Manufacturer: Raychem	Pressure (PSIA)	14.7	84.7	Section II.E	1	Test and Evaluation	
Model Number: Flamtrol	Relative Humidity (%)	100	100	Section II.E	1	Test and Evaluation	
Purchase Order Number: 34490-1601-71 and 87683-Q	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Electrical Control Cable	Radiation (Rad)	7.6×10^4	2×10^8	Section II.E	1	Test and Evaluation	
Accuracy: Spec: Demo: Location:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1 & Section IV	Test and Evaluation	
Electrical Penetration Room Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. Franklin Research Institute Test Report No F-C4033-1, dated January 1975.	1. Electrical cable with XLPE insulation and jacket: 2/C #12 AWG, 2/C #14 AWG, 3/C #14 AWG, 3/C #16 AWG, 6/C #14 AWG, 12/C #16 AWG, 2/C #16 AWG shielded, 2/C #16 AWG twisted, 2/C #10 AWG, 2/C #12 AWG and 3/C #10 AWG.

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Sheet 2

Component Electrical Cables

1. 600 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #14 AWG
   - 3/C #14 AWG
   - 3/C #16 AWG
   - 6/C #14 AWG
   - 12/C #16 AWG
   - 2/C #16 AWG Twisted and Shielded
   - 2/C #16 AWG Twisted

2. 1000 Volt Class Cables, Sizes as Detailed Below, With XLPE (Radiation Cross-Linked Polyolefin) Insulation and Jacketing Material

   - 2/C #12 AWG
   - 2/C #10 AWG
   - 2/C #12 AWG
   - 3/C #10 AWG

The subject cables are used in Class 1E circuits outside the containment in areas not subjected to submergence or water spray. The qualification is based on the following documents and discussion:

A. Document Reference 1 demonstrates that the cable with identical insulation is qualified for use inside the containment based on 40-year life and LOCA. In containment, environmental conditions are far more severe than the conditions outside the containment. The test cycle included the following:

   1. Thermal preaging at 150°C (302°F) for 25 days and at 160°C (320°F) for 12 days followed by thermal and radiation aging at 150°C (302°F) and 5×10^7 R for 7 days.

   2. Combined LOCA and radiation exposure as below:

      - Radiation 1.5×10^8 R.

      - Temperature and pressure as follows:

        177°C (351°F) at 84.7 psia for 10 hours; 135°C (275°F) at 45.7 psia for 4.5 days; 100°C (212°F) at 24.7 psia for 26 days.

      The chemical spray during LOCA consisted of 3,000 ppm of boron and 0.064 molar sodium thiosulfate and adjusted with sodium hydroxide to a pH value of between 9.5 to 11.

   3. Dielectric test for five minutes while immersed in water.

Sheet 2 (Contd)

It is concluded that the temperature, radiation, humidity and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for this application.

EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	90	346	Section II.E	1	Test and Evaluation	
Component: Note 1 Manufacturer: Rockbestos	Pressure (PSIA)	14.7	127.7	Section II.E	1	Test and Evaluation	
Model Number: Firewall III	Relative Humidity (%)	100	100	Section II.E	1	Test and Evaluation	
Purchase Order Number: -	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Electric Control Cable	Radiation (Rad)	1.9×10^5	2×10^8	Section II.E	1	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	1	Test and Evaluation	
Location: Sphere Ventilating Room Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation: N/A Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. Qualification of Firewall III Class 1E Electric Cables	1. Electrical Cable, 2/C #14 AWG XLPE/NEO

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Sheet 2

Component Cable Type 5

Type 5 - 2/C #14 AWG, XLPE/NEO, Firewall III, 600 V

A. The above cable is a replacement to the existing cables outside the containment in areas not subjected to submergence or water spray.

B. Document Reference 1 demonstrates that the cable with identical insulation is qualified (without taking any credit for the jacket) for use inside the containment based on 40-year life and LOCA. In containment, environmental conditions are far more severe than the conditions outside the containment. The test cycle included the following:

   1. Thermal aging at 150°C (302°F) for 1,300 hours (against the requirement of 850 hours to simulate a 40-year life); gamma radiation from a Cobalt-60 source for a total dosage of 2 x 10^8 rads; a LOCA simulation test with chemical spray at a peak temperature of 346°F; 100% relative humidity and a peak pressure of 113 psig. Total duration of test was 30 days.

   2. After completion of a LOCA test, further thermal aging at 200°F and 100% relative humidity for 100 days.

   3. Dielectric test for 5 minutes at 80 V ac/mil was performed after LOCA and thermal aging tests while immersed in water.

C. No credit has been taken for the neoprene jacket furnished. It is concluded that the temperature, radiation, humidity and aging requirements have been exceeded by a substantial margin and, therefore, the cable is qualified for use in this application.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	12 Months	Section II.D	2, 3	Evaluation	
Plant I.D. Number:	Temperature (°F)	235	250	Section II.F	2, 3	Evaluation	
Component: Note 1 Manufacturer: Section V	Pressure (PSIA)	41.7	41.7	Section II.E	Sheet 2	Evaluation	
Model Number:	Relative Humidity (%)	100	100	Section II.E	3	Evaluation	
Purchase Order Number: 3159-F, -1679, -1200	Spray	Lake Water	Sheet 2	Section II.E	3	Evaluation	
Function/Service: Electrical Power/Control Cable	Radiation (Rad)	7.3×10^5 Gamma 1.3×10^7 Beta	5×10^6 Gamma 1.3×10^7 Beta	Section II.E	1	Evaluation	
Accuracy: Spec: Demo: Location: Containment	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	2, 3 and Section IV	Evaluation	
Elevation: Some Assumed <590	Submergence	Yes	Sheet 2	Section II.E	Sheet 2 & 3	Evaluation	
Flood Level Elevation: 590 Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transactions paper by R B Blogett and R G Fisher on "Insulations and Jackets for Control and Power Cables in Thermal Reactor Nuclear Generating Stations," 11/23/68.	1. Electrical cable butyl insulation with PVC jacket.

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EQUIPMENT QUALIFICATION REPORT

Component Sheet No.:
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Date:

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Paper by N D Kenney, T N Mitropoulos and W L Seamonds on "Electrical Characteristics of Butyl Insulation at 125°C," Second Symposium on Butyl Rubber for Wire and Cable Insulations, AIEE Special Publication S138, June 1962. 3. Paper by J C Carroll and J R Maher on "Continued Evaluation of Butyl Rubber Insulated Cable," Second Symposium on Butyl Rubber for Wire and Cable Insulations, AIEE Special Publications S138, June 1962.	

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Sheet 2

Component Cable Types 21 and 24

Type 21 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 24 - 1/C #6 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 22 - 3 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

The subject cables are used in Class 1E circuits inside the containment. The qualification is based on the following documents and discussion:

A. Per Document Reference 1, the butyl insulation can withstand a radiation dose of up to 5×10^6 R with no change in dielectric property and 5×10^7 R with 20% reduction in dielectric property. The 30-day integrated radiation dose during accident is 7.3×10^5 R (gamma) and 1.3×10^7 (beta). The beta radiation decreases by an order of 10 for approximately 30 mil thickness of insulation. Thus, allowing for cable jacket, conduit and tray cover, etc, the total radiation dose at the cable insulation will be less than 5×10^6 R. This meets the environmental requirements on radiation.

B. Per Document Reference 2, cable with butyl insulation can withstand 125°C (257°F) for four weeks with no change in dielectric property. Also, the dielectric strength relatively remains constant after exposing to air oven aging test at 121°C (250°F) for 12 months.

C. Per Document Reference 2, the physical property of cable is within acceptable limits after exposing to air oven test at 121°C for 28 months continuously and for 60 months with alternate seven days at room temperature. This accelerated aging test parameters far exceed the 40-year life requirement for cable in an environment where the normal ambient is 80°F. This approach is based on ten degree half life rule per IEEE-101 (1974). Document Reference 3 also supplements this data where the test was done at 121°C for 293 weeks.

E. Per Document Reference 3, the cable passed the dielectric test after immersion in water for 160 weeks and, therefore, will be adequate in the 100% humidity and containment spray environment. Big Rock Point containment sprays are lake water and noncorrosive.

F. No credit has been taken for the jacket furnished.

G. There is no difference in pressure between any two terminations of a particular cable inside the containment and the effect of pressure on the ability of the cable to perform during an accident will not be impaired

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1	Evaluation	
Plant I.D. Number:	Temperature (°F)	235	245	Section II.E	1	Evaluation	
Component: Note 1	Pressure (PSIA)	41.7	Sheet 2	Section II.E	1	Evaluation	
Manufacturer: Section V	Relative Humidity (%)	100	100	Section II.E	1	Evaluation	
Model Number:	Spray	Lake Water	Sheet 2	Section II.E	1 ·	Evaluation	
Purchase Order Number: 3159-F-2034	Radiation (Rad)	7.3×10^5 Gamma 1.3×10^7 Beta	Sheet 2	Section II.E	1	Evaluation	
Function/Service: Electrical Power Cable	Aging	40 Years + LOCA	Sheet 2	Section II.E	1	Evaluation	
Accuracy: Spec: Demo:							
Location: Containment	Submergence	Subject To Submergence	Sheet 2	Section II.E	Sheet 2	Evaluation	
Elevation: Some Assumed <590							
Flood Level Elevation 590 Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transactions Paper 68TP651-PWR "Insulations and Jackets for Control and Power Cables in Thermal Reactor Nuclear Generating Stations," 4/23/68.	1. Electrical cables with PVC insulation and jacket - 3/C #12 AWG and 1/C #12 AWG.

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Sheet 2

Component Electrical Cables

600 Volt Cables With Sizes as Detailed Below With PVC (Polyvinyl Chloride) Insulation and Jacket

Type 29 - 3/C #12 AWG 7 Strand

Type 30 - 1/C #12 AWG 7 Strand

The above cables are used in Class 1E circuits and are qualified for use inside the containment as below:

A. Per Table 1 and Page 3 of Document Reference 1, the PVC insulation exhibits satisfactory physical properties (tensile strength and elongation) when exposed to gamma radiation up to 5×10^7 R. At 5×10^6 R, the tensile strength increases to 104% of its initial value (before radiation) and percent retention for elongation increases to 115%. When taking credit for the jacket, conduit and tray cover to absorb the beta radiation, this withstand value meets or exceeds the requirements.

B. From Page 4 and Table VI of Document Reference 1, PVC insulating material can retain (tested when immersed in water) a dielectric strength in excess of 86% of its initial value when subjected to 5×10^6 R and 72% when subjected to 5×10^7 R.

The subject 600 V cables are used in 125 V dc circuits and there is conservative margin in dielectric strength for continued operation even after exposure to a radiation dose of 5×10^7 R.

C. From Page 3 and Table II of Document Reference 1, the PVC can withstand a temperature of 200 hours at 136°C (245°F) and has a useful life in excess of 40 years. This tested value is in excess of the parameters required by ICEA (formerly IPCEA) standards for air oven test; ie, 121°C for 168 hours.

D. The peak temperature inside the containment of 235°F after an accident at Big Rock exists for less than one hour and decreases to 80°F in less than three days. The withstand temperature and duration exceeds the requirements.

E. There is no difference in pressure between any two terminations of a particular cable inside the containment and the effect of pressure on the ability of the cable to perform during an accident will not be impaired.

F. Per procurement documents, the subject cables were manufactured to meet ICEA (formerly IPCEA) standards including the accelerated water absorption test requi.ements.

Sheet 2 (Contd)

Summary

1. Paragraphs A, B and F provide evidence for radiation, humidity, water spray and submergence qualifications.

2. Paragraphs C, D and E provide evidence for temperature, pressure and aging qualifications.

Sheet 2 (Contd)

Summary

1. Paragraphs A, B and F provide evidence for radiation, humidity, water spray and submergence qualifications.

2. Paragraphs C, D and E provide evidence for temperature, pressure and aging qualifications.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	1	Test and Evaluation	
Plant I.D. Number: –	Temperature (°F)	235	Sheet 2	Section II.E	3	Test and Evaluation	
Component: Note 1 Manufacturer: Section V	Pressure (PSIA)	41.7	Sheet 2	Section II.E	Sheet 2	Evaluation	
Model Number: –	Relative Humidity (%)	100	Sheet 2	Section II.E	1	Test and Evaluation	
Purchase Order Number: Note 2	Spray	Lake Water	Sheet 2	Section II.E	Sheet 2	Evaluation	
Function/Service: Electrical Power and Control Cable	Radiation (Rad)	7.3×10^5 Gamma 1.3×10^7 Beta	Sheet 2	Section II.E	1, 2	Test and Evaluation	
Accuracy: Spec: Demo: Location: Containment	Aging	40 Years + LOCA	Sheet 2	Section II.E	1, 3	Test and Evaluation	
Elevation: Some Assumed <590	Submergence	Subject To Submergence					
Flood Level Elevation: 590 Above Flood Level: Yes: No: X							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transaction on Power Apparatus and Systems, May 1969. 2. "Fundamentals of Nuclear Hardening of Electrical Equipment" by L W Ricketts.	1. 600 V power and control cable, see Sheet 2. 2. PO #3159-F-1201, -2034, -1679, -2153, -2407, -1200.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
3. "Manual for Process Engineering Calculations" by Loyal Clarke and Robert L Davidson.	

Sheet 2

Component 600 Volt Power and Control Cable

1. Component: 600 volt power and control cable wi polyethylene insulation with polyvinyl chloride jacket. Cables are of the following types:

   Type 2 - 3/C #14 AWG

   Type 3 - 5/C #14 AWG

   Type 4 - 7/C #14 AWG

   Type 5 - 2/C #14 AWG

2. The above cables are used in Class 1E circuits inside the containment.

   The cable will be exposed to lake water spray and submergence.

2.1 The qualification information of the subject cable is based on the following documents and discussions:

2.1.1 TEMPERATURE

Document Reference 3, Chapter D, Table D-22, Page 71, indicates that the highest usable continuous temperature for polyethylene is 212°F.

Ambient temperature at Big Rock containment is only 90°F.

Engineering judgment shows that cable will be able to withstand 235°F for maximum period of one hour during LOCA.

This is based on the reasoning that the continuous withstand temperature is 212°F and a marginal increase in temperature to 235°F for one hour will only decrease its thermal life and not degrade its performance.

2.1.2 HUMIDITY AND SPRAY

Document Reference 1 on Page 535, Table IX, states that HD polyethylene does not show any signs of degradation when subjected to 90°C water for durations exceeding nine weeks. Based on this, it is concluded that the cable will be able to function satisfactorily under 100% humidity and water spray conditions. Big Rock Point containment spray water is noncorrosive lake water.

2.1.3 RADIATION

a. Document Reference 2, Section 3.3, Table 3.3, Page 121, indicates that there will be no "measurable effects" on polyethylene on exposure to radiation of 10^7 rads.

Sheet 2 (Contd)

b. Document Reference 1 on Page 535, Table IX, states that irradiated cable (5×10^7 R) can withstand immersion in 90°C water for more than nine weeks without electrical failure so radiation dose of 5×10^7 rads exceeds the total requirement of 1.373×10^7 rads (beta 1.3×10^7 R and gamma 7.3×10^5 R).

c. Cable jacket, trays and conduits provide shielding against beta radiation. Therefore, the conductor insulation will be exposed to a significantly lower dose of beta radiation.

2.1.4 PRESSURE

Review of the post-LOCA test reports for various types of cables indicates that pressure-related cable failures are extremely rare. Therefore, even though there is no test data available to prove pressure withstand, it appears very unlikely that cable failure can occur due to short duration peak pressure of the order of 41.7 psia.

2.1.5 TEST SPECIMEN AGING

a. Per Document Reference 1, Page 535, Table IX, irradiated (5×10^7 rads) cable can withstand immersion at 90°C water for more than nine weeks.

b. Document Reference 3, Chapter D, Table D-22, Page 71, indicates that the highest usable continuous temperature for polyethylene is 212°F.

Based on the above references, it would appear that the cable is not generally susceptible to significant radiation induced or heat induced aging degradation.

2.1.5 SUBMERGENCE

Document Reference 1, Page 535, Table IX, indicates that irradiated (5×10^7 rads) cable successfully passed dielectric tests after immersion in 90°C water for nine weeks.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	30 Days	Section II.D	2, 3	Evaluation	
Plant I.D. Number: -	Temperature (°F)	100	250	Section II.E	2, 3	Evaluation	
Component: Note 1 Manufacturer: Section V	Pressure (PSIA)	14.7	14.7	Section II.E	1, 2, 3	Evaluation	
Model Number: -	Relative Humidity (%)	80	100	Section II.E	3	Evaluation	
Purchase Order Number: 3159-F-1679 -1200	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Electrical Power and Control Cable	Radiation (Rad)	7.6×10^4	5×10^6	Section II.E	1 & Section IV	Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	40 Years + LOCA	Section II.E	Section IV & 2, 3	Evaluation	
Location: Electrical Penetration Room Elevation:	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation Above Flood Level: Yes: No:							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transactions paper by R B Blogett and R G Fisher on "Insulations and Jackets for Control and Power Cables in Thermal Reactor Nuclear Generating Stations."	1. Electrical cable with butyl insulations and PVC jacket.

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EQUIPMENT QUALIFICATION REPORT

Owner: Consumers Power Company
Facility: Big Rock Point
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Component Sheet No.:
Revision:
Date:

DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. AIEE Transactions paper by N D Kenney, T N Mitropoulos and W L Seamonds on "Electrical Characteristics of Butyl Insulation at 125°C." 3. AIEE Transactions paper by J C Carroll and J R Maher on "Continued Evaluation of Butyl Rubber Insulated Cable."	

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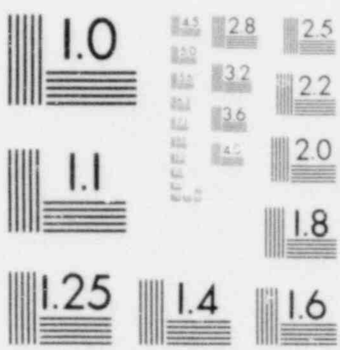
IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)



6"

MICROCOPY RESOLUTION TEST CHART

Sheet 2

Component Electrical Cables

Type 21 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 22 - 3 - 1/C #12 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

Type 24 - 1/C #6 AWG 7 Strand 600 V Cable With Butyl Insulation and PVC Jacket

The subject cables are used in Class 1E circuits outside the containment. The qualification is based on the following documents and discussion:

A. Per Page 534 in Document Reference 1, the butyl insulation can withstand radiation up to 5 x 10^6 R with no change in dielectric property. This exceeds the environmental requirements on radiation.

B. Per Document Reference 2, cable with butyl insulation can withstand 125°C (257°F) for four weeks with no change in dielectric property. Also, the dielectric strength relatively remains constant after exposing to air oven aging test at 121°C (250°F) for 12 months.

C. Per Document Reference 2, the physical property of cable is within acceptable limits after exposing to air oven test at 121°C for 28 months continuously and for 60 months with alternate seven days at room temperature. These accelerated aging test parameters far exceed the 40-year life requirement for cable in an environment where the normal ambient ranges from 5°C to 38°C (40°F to 100°F). This approach is based on ten degree half-life rule per IEEE-101 (1974). Document Reference 3 also supplements this data where the test was done at 121°C for 293 weeks.

D. Per Document Reference 3, the cable passed the dielectric test after immersion in water for 160 weeks.

E. No credit has been taken for the jacket furnished.

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EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REFERENCES		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	Sheet 2	Section II.D	1	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	235	Sheet 2	Section II.E	2	Test and Evaluation	
Component: Electrical Penetration Cable Manufacturer: General Electric	Pressure (PSIA)	41.7	Sheet 2	Section II.E	1, 3	Test and Evaluation	
Model Number: Note 1	Relative Humidity (%)	100	Sheet 2	Section II.E	1	Test and Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Electrical Power and Control Cable	Radiation (Rad)	7.3×10^5 gamma 1.3×10^7 beta	Sheet 2	Section II.F	1 and Section IV	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	1, 2	Test and Evaluation	
Location: Inside Penetration Elevation: Above 602'	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation 590 Above Flood Level: Yes X No							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transaction on Power Apparatus and Systems, May 1969.	1. Electrical Cable With Styrene Butadiene Rubber Insulation (Versotol) and Geoprene Jacket Used In Penetrations Types 1, 2, 4 & 7. Penetrations Are Discussed Elsewhere.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Manual for Process Engineering Calculations by Loyal Clarke and Robert L Davidson. 3. Final Integrity Test and Leakage Rate Determination of Big Rock Plant by Bechtel Corp, Job 3159, Dated July 1962.	

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Sheet 2

Component Electrical Cable

1.0 Component: (1) #14 AWG control cable GE Type RHW 75°C Versotol (styrene butadiene rubber) insulation and geoprene jacket. Versotol is also known as GR-S or BUNA S and (2) #8 AWG power cable GE Type RHW 75°C Versotol insulation and geoprene jacket.

2.0 The above cable is used in Class 1E penetration from inboard to outboard. The cable will not be exposed to containment spray or submergence.

2.1 The qualification information of the subject cable is based on the following documents and discussions:

2.1.1 Temperature

a. Document Reference 2, Figure D-19, which shows the high temperature limits of the various rubbers, indicates that BUNA S can withstand up to 290°F.

b. Document Reference 1 indicates that the irradiated (5×10^7 R) cable can withstand a combined high temperature and steam environment for four days at 40 psig and 142°C (287.6°F).

These values of 290°F and 287.6°F are higher than the 235°F (peak) for one hour required for the Big Rock Point Plant.

2.1.2 Humidity

Document Reference 1, Page 534, describes the LOCA test on irradiated cable specimens. The specimens were exposed to a radiation dose of 5×10^7 R and then placed in water-filled jars. The jars were inserted into a steam autoclave and subjected to a temperature of 142°C (288°F) and 40 psig (54 psia) for four days. Results of the test, tabulated on Table IX, Page 535, of Reference 1 indicates that the specimen with SBR insulation successfully passed dielectric test at 80 V/mil at 90°C (194°F). The BRP LOCA profile calls for a peak temperature of 235°F, 100% humidity and peak pressure of 41.7 psia for one hour, dropping to 140°F and 25 psia in 24 hours. After three days, temperature drops to the normal ambient of 80°F.

Because the test temperature, humidity, pressure and duration exceeded the BRP LOCA requirements, it is concluded that the cable is qualified to withstand the BRP LOCA environment.

2.1.3 Radiation

Document Reference 1 in Table XI indicates the threshold of gamma radiation damage (rad) for elastomer-based cable insulations. The

Sheet 2 (Contd)

table states that cable with SBR insulation can withstand doses up to 5×10^7 rad gamma.

It is concluded that the cable is qualified to Big Rock requirements based on the following evaluation:

a. Reference 1 indicates that the cable can withstand 5×10^7 rad gamma which exceeds the requirement of 7.3×10^5 R gamma and also exceeds the total requirement of 1.373×10^7 rads (1.3×10^7 R beta and 7.3×10^5 R gamma).

b. Cable jacket provides shielding against beta radiation. Therefore, the conductor insulation will be exposed to significantly lower doses of radiation.

2.1.4 Pressure

It can be concluded that the cable is qualified to Big Rock Point requirement based on the following:

a. Document Reference 3 includes a report on final integrity test and leakage rate determination for reactor containment of the Big Rock Point Plant. This report indicates a satisfactory leakage rate of 0.21% per 24 hours at 10 psig and further concludes that the leakage at design pressure of 27 psig would be in the same order as the leakage at 10 psig. The report states "it is safe to say that the leakage at design pressure is far below the value of 0.5% per day assumed in the analysis of maximum credible accident."

b. Document Reference 1 indicates that the irradiated (5×10^7 R) cable can withstand a combined high temperature, steam and pressure environment (40 psig or 54.7 psia) for four days.

Sheet 2 (Contd)

2.1.5 Test Specimen Aging

Document Reference 1 shows that the irradiated (5×10^7 R) cable can withstand a combined high temperature and steam environment for four days at 40 psig and 142°C (287.6°F). Even though there is no evidence that the test samples were preaged prior to steam test, the test proved the capability to withstand 142°C for at least four days.

Based on the normal containment ambient temperature of 90°F (32.2°C), using the 10°C one-half life rule, the duration of aging required to prove 40-year life is seven days at 143°C. Even though the test duration was four days only against the requirement of seven days, it is our judgment that the combined effect of high-pressure, steam and high temperature environment is far more severe than the anticipated in-containment environmental conditions following the worst case LOCA. The test pressure was 54.7 psia (40 psig) against the requirement of 41.7 psia and peak temperature during test was 289.6°F for four days against the requirement of 235°F maximum for one hour only.

Further, Document Reference 2 indicates that the maximum withstand temperature for SBR is 290°F which provides further credence to the conclusion that at relatively low temperatures such as 90°F (normal ambient) the cable should be capable of performing satisfactorily for 40 years.

The cable pigtails which have been in service for approximately 18 years were recently visually examined. It was noted that the conductor insulation did not exhibit any sign of aging related degradation such as cracks, brittleness, loss of flexibility, discoloration, etc, both inside and outside the containment. This is to be deemed as further evidence of aging qualification for the cable.

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	Parameter	Accident	Qualification	Accident	Qual		
System: Miscellaneous	Operating Time	30 Days	Sheet 2	Section II.D	1	Test and Evaluation	
Plant I.D. Number:	Temperature (°F)	235	Sheet 2	Section II.E	2	Test and Evaluation	
Component: Electrical Penetration Cable; Manufacturer: Okonite	Pressure (PSIA)	41.7	Sheet 2	Section II.E	1, 3	Test and Evaluation	
Model Number: Note 1	Relative Humidity (%)	100	Sheet 2	Section II.E	1	Test and Evaluation	
Purchase Order Number:	Spray	Not Subject To Spray	-	-	-	-	
Function/Service: Electrical Power Cable	Radiation (Rad)	7.3×10^5 gamma 1.3×10^7 beta	Sheet 2	Section II.E	1	Test and Evaluation	
Accuracy: Spec: Demo:	Aging	40 Years + LOCA	Sheet 2	Section II.E	Section IV 1, 2	Test and Evaluation	
Location: Inside Penetrations; Elevation: Above 602'	Submergence	Not Subject To Submergence	-	-	-	-	
Flood Level Elevation 590; Above Flood Level: Yes: X No:							

DOCUMENTATION REFERENCES	NOTES
1. IEEE Transactions on Power Apparatus and Systems, May 1969.	1. 3 kV Electrical Cable With Oil Base Rubber Insulation and Neoprene Jacket Used in Type 5 Penetrations. Separate Qualification Reports for the Penetration Assemblies Are Elsewhere in This Report.

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DOCUMENTATION REFERENCES (cont)	NOTES (cont)
2. Manual for Process Engineering Calculations by Loyal Clarke & Robert L Davidson. 3. Final Integrity Test and Leakage Rate Determination of Big Rock Plant by Bechtel Corp, San Francisco, CA, Job 3159, Dated July 1962.	

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Sheet 2

Component 3 kV Power Cable

1.0 Component: 3 kV power cable.

1.1 Single conductor #4/0 AWG nonshielded ozone resisting oil-based rubber insulation, heavy-duty black neoprene jacket, Okonite cable.

2.0 The above cable is used in Class 1E penetration from inboard to outboard. This cable will not be exposed to containment spray or submergence.

2.1 The qualification information of the subject cable is based on the following documents and discussions:

2.1.1 Temperature

a. Document Reference 2, Figure D-19, shows the high temperature limits of the various rubbers. The above referenced figure shows that natural rubber can withstand up to 270°F. It is an established fact that oil-based rubber has better heat withstand capability than natural rubber.

b. Document Reference 1 indicates that cable did not have any deteriorating effect when irradiated (5×10^7 R). Cable was subjected to high temperature of 287.6°F and steam environment for 25 days.

These values of 270°F and 287.6°F are higher than the 235°F required for Big Rock Point.

2.1.2 Humidity

Document Reference 1, Page 534, states that irradiated cable sample was subjected to conditions simulating the steam environment expected within the containment vessel. Water-filled jars containing the sample were maintained in a steam autoclave at 40 psig (142°C) for maximum period of 32 days. Table IX in Reference 1, IEEE Transaction, states that irradiated cable (5×10^7 R) can withstand a combined high temperature and steam environment for 25 days.

It is, therefore, concluded that the cable can function satisfactorily in 100% humidity environment.

2.1.3 Radiation

Document Reference 1 in Table XI states threshold of gamma radiation damage (rad) for elastomer-based cable insulations. The table states that cable with 90°C oil-base insulation can withstand 10^8 rad gamma radiation.

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It is concluded that the cable is qualified to Big Rock Point requirements based on the following:

a. Reference 1 indicates that the cable can withstand 10^8 rad gamma which exceeds the requirement of 7.3×10^5 R gamma and also exceeds the total requirement of 1.373×10^7 R (1.3×10^7 R beta and 7.3×10^5 R gamma).

b. Cable jacket provides shielding against beta radiation. Therefore, the conductor insulation will be exposed to a significantly lower dose of radiation.

2.1.4 Pressure

It can be concluded that the cable is qualified to Big Rock Point requirement based on the following:

a. Document Reference 3 includes a report on final integrity test and leakage rate determination for reactor containment of the Big Rock Point Plant. This report indicates a satisfactory leakage rate of 0.21% per 24 hours at 10 psig and further concludes that the leakage at design pressure of 27 psig would be in the same order as the leakage at 10 psig. The report states "it is safe to say that the leakage at design pressure is far below the value of 0.5% per day assumed in the analysis of maximum credible accident."

b. Document Reference 1 indicates that irradiated (5×10^7 R) cable can withstand a combined high temperature steam and pressure environment for 25 days at 40 psig (54.7 psia) and 142°C.

2.1.5 Test Specimen Aging

Based on the following evaluation, it is concluded that the cable is qualified for 40-years plus LOCA aging:

Document Reference 1 in Table II indicates the estimated life of oil-base insulation is 61 years at 70°C (168°F) which is far above the value specified for the Big Rock Point Plant.