BOSTON EDISON COMPANY
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BOSTON, MASSACHUSETTS 02199

WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

August 3, 1984
BECo Ltr. No. 84.119

Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing
Office Of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

License No. DPR - 35
Docket No. 50-293

SUBJECT: Environmental Qualification of Safety-Related Electrical Equipment at Pilgrim Nuclear Power Station

REFERENCES: 1) BECo Letter No. 84-099 dated 7/9/84, W. D. Harrington to D. B. Vassallo
2) Meeting between BECo and the NRC on May 22, 1984

Dear Sir:

Reference (1) provided you with Boston Edison's resolution for each of the equipment covered under the Technical Evaluation Report (TER) written by Franklin Research Center. For the equipment items that have been added to the "Master List of Electrical Equipment" and not factored in the TER resolution process, Boston Edison stated that it would submit a response with resolution and applicable JCO's on August 3, 1984.

Enclosure 1 to this letter describes the current resolution for each of the added equipment, in a matrix format. Enclosure 2 provides you with a justification for continued operation (JCO) for any equipment for which the documentation for environmental qualification is not yet completed. Enclosure 2 also includes JCO's for some TER items which were not included in Enclosure 2 of Reference 1.

Your Staff has requested additional information regarding compliance with four areas in 10CFR 50.49. These four areas are:

(1) JCO's;
(2) 50.49 (b) (2) review Methodology
(3) DBE's including flooding outside containment; and
(4) Regulatory Guide 1.97 equipment

Boston Edison has stated its approach with respect to these four areas in Reference (1) and reiterates the following to respond to the request for additional information.

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(1) JCO's
Based on the JCO's submitted in Enclosure 2 of Reference 1 and in Enclosure 2 of this letter Boston Edison states that no significant degradation of required safety functions is expected to occur nor is operator confusion expected to inhibit the accomplishment of required safety functions due to failure of equipment under design basis accident environments.

(2) 50.49 (b) (2) Review Methodology
In performing its review of the methodology to identify equipment within the scope of 10CFR 50.49 (b) (2), Boston Edison has performed a series of studies in response to I&E Information Notice 79-22, I.E. Bulletin 79-27 and a review of associated circuits performed under the auspices of Appendix R. The findings of these evaluations conclude that there is no equipment identified under 50.49 (b) (2). Nevertheless, Boston Edison intends to verify this assessment using the guidelines provided by your staff. Any deviation identified as a result of this assessment will be factored into the BECo Environmental Qualification Program and reported in accordance with PNPS Technical Specification reporting procedures.

(3) DBEs including flooding outside containment
As part of the effort in identifying the Master List of equipment within the scope of 10CFR 50.49 (b) (1), Boston Edison has reviewed all postulated design basis events documented in the FSAR including a Loss Of Coolant Accident (LOCA) inside containment and High Energy Line Break (HELB) outside containment, including flooding outside containment.

(4) Regulatory Guide 1.97 equipment
The equipment within the scope of 10CFR 50.49 (b) (3) is all R. G. 1.97 Category 1 and 2 equipment and will be identified in the R. G. 1.97 submittal. After staff review and acceptance of R. G. 1.97 submittal, Boston Edison will add the applicable equipment to the E. Q. Master Equipment List and will implement its schedule of R. G. 1.97 activities including environmental qualification.

As stated in Reference 1, it is requested that supplemental SER's be issued incorporating your review of Reference 1 as well as this submittal. We would be pleased to answer any questions you may have regarding this submittal or the information contained in Reference 1.

Very truly yours,

A L Oxsen for

W. D. Harrington

Enclosure 1: Resolution Matrix for new equipment.
Enclosure 2: Justification for Continued Operation.

TAV/mm

ENCLOSURE-1

PILGRIM NUCLEAR POWER STATION - S.E.R. DFFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
1, 2, 3, 4b, 4c, 9, 14, 16, 17, 18, 19, 22a, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 35, 36, 37, 38, 39, 40, 41	Motor Operator Limitorque/SMB	Aging degradation, qualified life	Inspection and replace component parts with qualified parts
4a, 5, 6, 10, 11, 12, 13, 15, 20, 21, 22b, 34	Motor Operator Limitorque/SMB	Aging degradation Qualified life Similarity Radiation	Replace with qualified motor operator - Limitorque
7, 8, 97, 256, 258, 260, 261	Standby Gas Treatment System Damper--Honeywell/M940A10671 Humidity Detectors Honeywell/R7088C Honeywell/Q464A Temp. Switch: Fenwall/40102010115 Transformer-GE/9T55Y46G7 Contractor--Allen Bradley/ 702LT0D93	Inadequate documentation	Design modification to establish qualification
259, 262	Standby Gas Treatment System Cable - Bronco 66	Inadequate Documentation	Replace with qualified cable--Vulkene Supreme or equivalent
95	Standby Gas Treatment System Heater - Chromalox/64-47499	None	Qualified Report 47066-HT-1
45a, 45f, 50, 53, 55, 56, 58, 59, 60, 61, 62a, 62b, 62c, 62d, 64, 65, 66, 67, 70, 73, 74, 75, 77, 78a, 78b, 78c, 78d, 79, 82	Solenoid Valves ASCO/NP8320A184E	Qualified life	Qualified: Test Report AQS21678/TR Qualified life determined by Analysis Report 47066-SOV-2.
85, 86	Solenoid Operator AVCO/C5159	Similarity Qualified life Functional testing	Negotiating to join testing program already in progress Est. completion date 1/85
49, 48	Solenoid Valves ASCO/HVA-90-405-2A /WP-LB-831636	Inadequate documentation Aging degradation Qualified life	Replace with qualified solenoid valves--ASCO NP8316

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
87, 91, 93, 94	Motors GE 5K6339XC87A 5K254AK299W1A 5K6337XC93A 5K184AL217	Inadequate documentation	Qualified: Test Report G-HK-0-16 Analysis Report 47066-MOT-3.1 Motor Terminations - Evaluate/ Replace
54, 57a, 57b, 57c, 57d, 72	Solenoid Valves Valcor/V526529231	Qualified life	Qualified: Test Reports QR52600-5940-2 QR52600-515 Qualified life established in analysis report 47066-SOV-8
81	Solenoid Operator Target Rock/1/2SMSA01	Inadequate documentation	Qualified: Test Report 2199A; Analysis Report 47066-SOV-6
233, 234, 235, 236, 237, 238, 239a/b/c, 240, 241a, 242	Cable Kerite/FR/FR, HT/FR, HT/NS	None	Qualified: Test Report 17446-2 and Analysis Report 47066-CAB-3
268, 269, 109, 107a/b/c/e, 108a/c	Indicating Light GE/ET-16 Switch GE/CR-2940 Relays: Johnson/SER KZ4000B Agastat/2412AN	Exempt	No active safety-related function. Components will be tested or replaced, when qualified replacement items are determined.
117	Cable Rockbestos/Firewall III	Inadequate documentation	Qualified: Test Rpts 2806, QR-1806, 110-11516, F-C-3798, F-C-5022-2 and Analysis Report 47066-CAB-5

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
243, 244, 245, 246, 247, 248	Cable Okonite/Okolon & Okoprene	None	Qualified: Test Report NQRN-1
110, 111, 112, 118, 119, 120, 121, 122, 123, 124,	Instrument Rack Wiring from J. B. to devices	Inadequate documentation	Replace with qualified equipment. Vulkene Supreme or equivalent
100	Ring Tongue Terminations Less Than 4KV in the Drywell	Inadequate documentation	Replace some terminations with qualified splices (Raychem WCSF-N). Where ring-tongues have been tested, verify installation adequacy.
252	Cable Electrical/Distribution Type SI	Inadequate documentation	Test program to be initiated 9/84 with completion expected by 3/85
265b/d/e/f, 267a/c	Terminal Block GF/EB-25	None	Qualified: Test Report QSR-010-A-01 & B0119
88, 89, 90	Motor Control Centers Cutler Hammer/6AF685046 Nelson Electric/1035E	Inadequate documentation Aging degradation Qualified life Similarity Radiation Test sequence	Design modification to enclose MCC's eliminating humidity, temperature and pressure effects. Analysis to address radiation in progress
92	Motor Louis Allis/COG4B	Inadequate documentation	Replace with qualified motors Westinghouse motors purchased from Buffalo-Forge using the DO-146F Qual. Report.
99	4KV Terminations Kerite	Inadequate documentation	Qualified: Test Reports F-C-4020-1 & F-C-4020-2. Qualified life evaluation. To be complete by 9/84

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
101, 102	Splices Raychem/WCSF-N	Aging degradation Qualified life	Qualified: Test Report 58442-1, Qualified life Analysis Report 47066-SPL-1.1
103, 104a, 104f, 104g, 104h, 104I	Terminal Blocks Buchanan/525	Inadequate documentation, similarity	Design modification to delete terminal blocks - Replace with qualified splice (Raychem WCSF-N)
210, 211, 212, 213, 214, 226, 227	Level Switch Yarway/4418C & 4418EC	Inadequate documentation	Replace component parts with qualified component parts (Yarway Kit #959552)
98, 263	Accelerometer TEC/ND	Inadequate documentation Qualified life	Qualified: Test Report 517-TR-03. Analysis Report 47066-MON-2
220, 221	Transmitter GE/555	Inadequate documentation	Replace with qualified transmitter - Rosemount 1153 Transmitter
232	Level Switch Robertshaw/SL702A1	Inadequate documentation	Required for radiation only--pending vendors material list
127, 129a, 129b, 129c, 129d, 128h, 128j	Electrical Penetrations GE/238X60NLG	Inadequate documentation, similarity	Qualified: GE prototype Test Report - Analysis Report 47066-PEN-1
132, 137	Radiation Detector GE/237X731G009	Inadequate documentation	Qualified: Test Report 943-81-003 and analysis report 47066-RAD-2

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
223	Transmitter Rosemount/1152	Aging degradation Qualified life	Qualified: Test Report 117415 Rev. B, Analysis Report 47066-PT-1 establishes qualified life. Installing Conax ECSA Conduit Seal.
139, 140, 142, 144, 143, 145, 147, 159, 160, 161, 162, 163, 164, 166	Temperature Switch Fenwall/17023 & 17002	Aging degradation Pressure Steam exposure Profile Functional testing	Qualified by existing Test Report BECo is negotiating to obtain the rights for its use
171, 174, 175, 177, 178, 179, 206, 222	Pressure & Differential Pressure Switch Barton/288, 288a, 289a	Aging degradation, qualified life, similarity, temperature, pressure, radiation	Qualified: Test Reports; 145C3008, 145C3009, R3-288a-1. Analysis Report 47066-PS-2
173, 176, 180	Pressure & Diff. Pressure Switches Barton 288, 288a 289a	Aging degradation, qualified life, similarity, temperature, pressure, radiation	Replace with qualified equipment. Static-O-Rings
189, 190, 191, 192, 193, 197, 198, 202, 203, 204, 205	Pressure Switch Static-O-Ring/12N	Inadequate documentation Aging degradation Pressure Radiation	Test Report: 30203-2. Completion pending vendor's material list
181, 182, 208, 209	Pressure Switch Static-O-Ring/5N	Inadequate documentation Aging degration Temperature Pressure	Replace with qualified equipment. Static-O-Ring Model NO. 6N6.
183, 186, 187, 188, 199, 200, 201	Pressure Switch Barksdale/B2T	Inadequate documentation	Qualified: Test Reports 596-0398 & 15566-23 and Analysis Report 47066-PS-3
194, 196, 207	Pressure Switch Barksdale/B2T, D2H, P1H	Inadequate documentation Qualified life Steam exposure (profile) Radiation	Replace with qualified Equipment: Static-O-Ring.

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
195	Pressure Switch Mercoid/DA23804	Inadequate documentation	Replace with qualified pressure switch. Static-O-Ring Model 4N6
146	Temperature Element Thermo Electric/3544710	Inadequate documentation	Replace with qualified equipment. Weed RTD's Model No.SP-612D.
42, 152, 153, 154, 155, 156, 157, 158, 185	HPCI Turbine Controls Various Equipment	Inadequate documentation	Radiation only. Plant modification to address radiation
172	Pressure Switch Barton/278	Inadequate documentation	Test Report R3-288a-1. Replace component parts with qualified parts. Barton 288A Instrument Case.
249	Cable GE/Vulkene supreme	Inadequate documentation, similarity	Qualified: Test Report FC-4497-2 Analysis Report 47066-CAB-1.1
250	Cable GE/Vulkene SIS	Qualified	Test planned: To be initiated by 9/84 planned with completion by 3/85
251	Cable BIW/Bostrad	None	Qualified Test Report B901A

PILGRIM NUCLEAR POWER STATION - S.C.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
254, 255	Limit Switch Namco/EA740	Similarity	Replace with qualified equipment. NAMCO EA740 with EC210 Connector Assembly.
264b/d/e/f, 266a/c	Switch Electro Switch/24/40	Inadequate documentation	Test Reports: 2392-2, 2392-14, 3030-1 Switches will be tested or replaced when qualified replacements are determined
270	Cable GE/Vulkene SIS	Inadequate documentation, similarity	Qualified: Test Reports: 43905-2 & EPAQ-047
271, 272, 273, 274, 275, 276, 277, 278, 279, 280	Terminal Block GE/CR-151	None	Qualified: Test Reports: GEN-8-18 & BO119
43	Solenoid Valve Atkomatic/247214	Exempt	Radiation only - completion pending vendor's material list

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
44	Solenoid Valve Atkomatic/247214	Exempt	Out of Scope of 10CFR50.49
45b, 45c, 45d, 45e, 45g, 45h 45i, 46, 47, 62e, 78e, 78f, 80, 83, 84, 282	Solenoid Valve ASCO/NP8320A184E	Qualified life	Out of scope of 10CFR50.49
51	Solenoid Valve ASCO/HVA90405	Inadequate documentation	Out of scope of 10CFR50.49
52, 57E, 57F, 57G, 57H	Solenoid Valve Valcor/V5265683 /V526529231	Qualified life	Out of scope of 10CFR50.49
63, 68, 69 71	Solenoid Valve Valvor/V526529212	Qualified life	Out of scope of 10CFR50.49
76	Solenoid Valve ASCO/HT8210C22	Inadequate documentation	Out of scope of 10CFR50.49
104b, 104c, 104d, 104e,	Terminal Block Buchanan/525	Inadequate documentation similarity	Out of scope of 10CFR50.49
125	Electrical Penetration Conax/Modular Type	Inadequate documentation similarity	Out of scope of 10CFR50.49
224, 225	Level Switch Yarway/4418EC	Inadequate documentation	Out of scope of 10CFR50.49
281	Switch Electro Switch/24/40	Inadequate documentation	Out of scope of 10CFR50.49
107f/d, 108b	Indicating Light GE/ET-16	Exempt	Out of scope of 10CFR50.49

PILGRIM NUCLEAR POWER STATION -, S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
136	Transmitter Rosemount/1152	Aging Degradation Qualified Life	Out of scope of 10CFR50.49
264a/c, 266a	Electrical Switch Electro-Switch/24/40	Inadequate Documentation	Out of scope of 10CFR50.49
265a/c, 267b	Terminal Block GE/EB-25	None	Out of scope of 10CFR50.49
239D	Cable Kerite/HT-FR&HT-NS	None	Out of Scope of 10CFR50.49

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
126	Electrical Penetration Physical Science/Canister Type	Documentation Not Available	Out of scope of 10CFR50.49
129e, 128a/b/c/d/e/f/h/i	Electrical Penetration GE/238X60NLG	Inadequate documentation similarity	Out of scope of 10CFR50.49
130	Pressure Switch Meletron/92416SS5A	Inadequate documentation	Out of scope of 10CFR50.49
131, 133, 134, 168, 169, 216, 217	Transmitter GE/551	Inadequate documentation or exempt	Out of scope of 10CFR50.49
138	Transmitter Foxboro/611DM	Inadequate documentation	Out of scope of 10CFR50.49
148	Limit Switch NAMCO/EA740	Similarity	Out of scope of 10CFR50.49
149	Limit Switch NAMCO/D1200G2	Inadequate documentation	Out of scope of 10CFR50.49
151	Fuse Panel GE/238X278G1	Exempt	Out of scope of 10CFR50.49
165	Electric Heater	Inadequate documentation	Out of scope of 10CFR50.49

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

TER #	EQUIPMENT TYPE MANUFACTURER/MODEL #	TER DEFICIENCY	RESOLUTION
215	Level Switch	Inadequate documentation	Out of scope of 10CFR50.49
228	Level Switch McDonnel/63SY	Exempt	Out of scope of 10CFR50.49
229, 230, 231	Level Switch Robertshaw/SL305E7X /SL702A1	Exempt	Out of scope of 10CFR50.49
141	Thermostat Johnson Controls	Inadequate documentation	Out of scope of 10CFR50.49
184	Pressure Switch Mercoid/AP7021153	Inadequate documentation	Out of scope of 10CFR50.49
135	Temperature Element	Inadequate documentation	Out of scope of 10CFR50.49
150	Hydrogen Analyzer Comsip Delphi/K1Y	Aging degradation Qualified life Radiation	Out of scope of 10CFR50.49
114, 115, 116	Cable Rockbestos/Firewall III	Inadequate documentation	Out of scope of 10CFR50.49
257	Temperature Switch Fenwall/180230	Inadequate documentation	Out of scope of 10CFR50.49
253, 113	Indicating Light GE/ET-16 Terminal Block GE/EB-25	Exempt	Out of scope of 10CFR50.49

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolution
SVOS-1/2/3/4	Limit Switch NAMCO/SL3	Qualified: Radiation Only - Analysis Report 47066-LSW-3
LS302 - 82A/B/C/D	Level Switch ROBERTSHAW/SL-305-E7X	Replace with qualified equipment. System comprised of Rosemount 1153 Transmitter and Fluid Component Inc. FR72-4HTRDLL Heated RTD.
FT1049A/B	Transmitter GE/555	Replace with Qualified equipment, Rosemount 1153 Transmitter.
MO1301-62, MO1301-49 MO1001-19, MO2301-6, MO4065, MO4009A/B, MO4083, MO3806, MO3805	Motor Operator Limitorque/SMB	Inspect and replace component parts with qualified parts.
PS4008	Pressure Switch Barton/288A	Evaluation in progress

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolution
PS504A/B/C/D PS503A/B/C/D	Pressure Switch Barksdale/B2T, DIT	Qualified: Test Reports 596-0398 & 15566-23. Analysis Report 47066 - PS-3
Tip Shear & Ball Valves: 1, 2, 3, 4	Valves G.E.	Analysis in progress for radiation only requirement.
RE1735 A/B/C/D	Radiation Monitor GE/194X927	Qualified Test Report: QSR-015-A-01 Analysis Report 47066-RAD-1.1
RE1734A/B/C/D	Radiation Monitor GE/237X731	Qualified: Test Report 943-81-003 and analysis Report 47066-RAD-2.
B14	Motor Control Center Nelson Electric/1035E	Plant modification to eliminate accident environment

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolutions
Cable Codes: CX2, CX4, 520 S48, S37, CXG, CX8, C02, C03, S19, S27, Z3, Z3A	Cable	Currently being evaluated. Similarity analysis under evaluation. Some documentation complete. At the end of the analysis all undocumented equipment will be tested/replaced.
SVL61, SVL82, SVL83	Solenoid Valve ASCO/HT8320A107, HT8320A22	Replace with qualified equipment. ASCO NP8320
P202D/E/F	Motor GE/5K Series	Motor Qualified: Test Report: G-HK-0-163. Analysis Report 47066-MOT-3.1. Replace Terminations with Qualified Kerite or Raychem Terminations - Evaluate P202D
512A, 210A, 712A, 912A	Cable Okonite/Okolon	Qualified: Test Report NQRN-1 Analysis Report 47066-CAB-4.1
512B	Cable Rockbestos/Firewall III	Qualified: Test Report 2806 & QR-1806 & 110-11516 & F-C-3798 & F-C-5022-2 Analysis Report 47066-CAB-5
B9	Cable Okonite/Okoprene or Kerite/FR/FR	Qualified: Test Report NQRN-1, 17446-2 & 47066-CAB-3 Analysis Reports: 47066-CAB-4.1 and 47066-CAB-3
412A, 106, 212B, SC16	Cable Various Manufacturers	Currently being evaluated.

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolution
J720, J32, J217, J451, J866, C2257B, C2207A, J599, J600, J601, J602, J606, J444, J463, J561, J462, J603, J604, J874, J466	Terminal Block Unknown Manufacturers	Replace with qualified equipment.
J264, J409, J57, J58, J59, J300, J204, J205, J206, J207	Terminal Block Unknown Manufacturer	Replace with qualified splice - Raychem WCSF-N.
J317, J318, J256, J257, J258, J53, J54, J315, J316, J31, J33, J34, J51, J52, J562, J552, J456, J177	Terminal Block Buchanan/525/222	Qualified: Test Report B0119 and Analysis Report 47066-TB-1.
J859	Cable Splice Kerite	Evaluation in progress.
N912, N923, N921, CS42-1821, CS42-1822	Control Switch GE/CR2940	Evaluation in progress.
C61 A & B	Indicating Lights GE/CR2940UC	Evaluation in progress.
C129A & B, C2257A, C2207B	Instrument Rack Wiring and Terminal Blocks Unknown Manufacturer	Replace with Qualified wire and terminal blocks
J553	Terminal Block Buchanan/BiXX	Evaluation in progress.

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolution
P229	Motor Baldor	Evaluation in progress.
PS2390A & B	Pressure Switch Static-O-Ring/6N	Evaluation in progress.
C118, C119, J411, J412, J660, J661, J330, J523, J782, J681, J522, J684, J781, J787, J329	H_2/O_2 Analyzer	Replace with qualified equipment. Compsi Delphi
C68A & B, C69A & B	Control Panels Various Equipment	Plant modification to remove from scope of 10CFR50.49.
C2303, T2303	Local Control Panels Various Equipment	Under evaluation plant modification to establish qualification under consideration.
C2204, C2222	Local Control Panel Various Equipment	Evaluation in progress.
C2201, C2205A, C2260, J538 J539	Instrument Wire Manufacturer Unknown	Replace with qualified wire.
J623, J624, J625, J626, C2202	Unknown Contents	Walkdown and evaluation in progress

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY RESOLUTIONS

PNPS ID #	Equipment Type Manufacturer/Model #	Resolution
J863	Instrument Wire and Terminal Block Manufacturer Unknown	Replace with qualified equipment.
C150	Control Switch & Indicating Lights Electroswitch Series 40 GE ET-16 Lights	Evaluation in progress. Plant modification to establish mild environment under consideration
C2259, C2262, C2252A	Terminal Block GE/CR-151	Qualified: Test Report B0119 and Analysis Report 47066-TB-1.
Terminations 4KV	Ring Tong Connectors Various Manufacturers	Conduct Analysis/Undocumented equipment will be tested or be replaced

ENCLOSURE-2

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Motor Termination Splices - VAC201A, B; VEX210A, B
TER No. 91, 93 Sheet 1 of 1

Preparer:	JL Rogers	Date:	8/2/84
Independent Review:	MR Ern	Date:	8/2/84
Approval:	RC McDenny	Date:	8/2/84

The HPCI Area Cooling (VAC201A, B) and the Standby Gas Treatment System Fan motor termination splices are tape type motor termination splices with a glyptal outer covering. These splices are similar to the splices tested in FIRL Report #F-C3056.

Temperature and Pressure

The test splices were subjected to a steam environment for 7 1/2 days. The splices were electrically loaded throughout the exposure period. The peak temperature and pressure were 325°F and 80 PSIG for a duration of 16 hours. The temperature and pressure were reduced to 220°F and 5 PSIG for the remainder of the test. The samples were subjected to a chemical spray solution of 1900 ppm boron throughout the test.

Operability Time

The test time was 7 1/2 days. A degradation equivalency analysis shows that the test profile is thermally more degrading than the composite outside containment profile at PNPS for 30 days.

Aging

An aging analysis shows that the materials of the tested splice have a minimum expected life at 105°F of 272 years (glass tape). The expected life of the glyptal (assume alkyd varnish) at 105°F is greater than 1×10^5 years.

Radiation

A radiation analysis shows that the materials of the tested splice have a minimum threshold value of 1.3E6 rads gamma (silicon rubber tape). Per REIC Report No. 21, the dielectric properties of silicones are little affected unless absorbed doses exceed 2×10^8 rads gamma.

Although the insulation resistance decreased during the test, all of the cable splices remained functional throughout the test. Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1001-90A/D
TER No. 189b,c (A, C), 203a, b (B, D) Sheet 1 of 1

Preparer: JL Rogers Date: 7/23/84

Independent Review: WS Clancy Date: 7/24/84

Approval: [signature] Date: 7/26/84

The function of these pressure switches is to provide a high drywell pressure permissive to start the RHR and Core Spray pumps. These switches will be exposed to a harsh steam and radiation environment following a PBOC-2B and 2I (Reactor Water Cleanup System Pipe Breaks) and a harsh radiation environment following a PBIC and all other PBOC's.

From FSAR Appendix G, high drywell pressure does not result from a PBOC. Therefore, actuation of these switches to mitigate the effects of a PBOC is not required.

For PBIC's, radiation levels of 1×10^6 rads are not reached until at least 100 hours after the pipe break occurs. As per GE letter the limiting materials are neoprene and Buna-N. Per D.O.R. Guidelines, Buna-N has a radiation threshold of 1×10^6 rads and neoprene a radiation threshold of 1×10^7 rads. Therefore, switch failure due to radiation would not be expected to occur until at least 100 hours after the pipe break. Automatic start of the RHR and Core Spray pumps would not be required at this time.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS-512A/D
TER No. 190 (A, B), 202 (C, D) Sheet 1 of 1

Preparer: [signature] Date: 7-23-84

Independent Review: W A Clancy Date: 7/27/84

Approval: [signature] Date: 7/27/84

The function of these pressure switches is to provide a scram signal to the Reactor Protection System and to isolate Secondary Containment upon indication of high drywell pressure. These pressure switches are exposed to a harsh steam and radiation environment following PBOC-2B and 2T (Reactor Water Cleanup System Breaks) and a harsh radiation environment following a PBIC and all other PBOC's.

According to FSAR Appendix G, PBOC's do not produce high drywell pressure. Furthermore, subsequent failures of these pressure switches in the harsh environment caused by these PBOC's will not reverse the previously accomplished safety functions of scram and secondary containment isolation. Therefore, these switches do not need to be qualified for the effects of PBOC's.

For PBIC's, radiation levels of 1×10^6 rads are not reached until at least 100 hours after the pipe break occurs. As per GE letter the limiting materials are neoprene and Buna-N. Per D.O.R. Guidelines, Buna-N has a radiation threshold of 1×10^6 rads and neoprene a radiation threshold of 1×10^7 rads. Therefore, switch failure due to radiation would not be expected to occur until at least 100 hours after the pipe break. It can reasonably be assumed that the scram and secondary containment isolation functions would have been completed prior to this time.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1001-89A/D
TER No. 191 (A), 205 (B), 189a (C), 197 (D)

Preparer: J L Rogers Date: 8/1/84

Independent Review: MR Ervin Date: 8/1/84

Approval: RAlDenny Date: 8/1/84

The function of these pressure switches is to provide a high drywell pressure permissive to the ADS initiation logic. These switches are exposed to a harsh steam and radiation environment following PBOC -2B and 2T (Reactor Water Cleanup System Pipe Breaks) and a harsh radiation environment following a PBIC and all other PBOC's.

According to FSAR Appendix G, PBOC's do not produce high drywell pressure. Operator action is credited in the PNPS Emergency Operating Procedures to initiate ADS if required. Subsequent failure of these switches caused by a harsh environment will not prevent manual operation of ADS from the control room. Therefore, these switches do not need to be qualified for the effects of PBOC's.

For PBIC's, radiation levels of 1×10^6 rads are not reached until at least 100 hours after the pipe break occurs. As per GE letter the limiting materials are neoprene and Buna-N. Per D.O.R. Guidelines, Buna-N has a radiation threshold of 1×10^6 rads and neoprene a radiation threshold of 1×10^7 rads. Therefore, switch failure due to radiation would not be expected to occur until at least 100 hours after the pipe break. It can reasonably be assumed that an operator could actuate ADS if it would be required at this time.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1001-83A/D
TER No. 192 (A), 193 (C), 198 (D), 204 (B)

Preparer: Jc Rogers Date: 7-23-84

Independent Review: W S Cleary Date: 7/24/84

Approval: RA Cleary Date: 7/27/84

These pressure switches provide a drywell pressure permissive to the control logic of RHR valves 1001-23A/B, 1001-26A/B and 1001-37A/B. These valves must open in order to provide drywell and suppression pool spray for the purpose of primary containment cooling. These pressure switches may be exposed to a harsh steam and radiation environment following PBOC-2B and 2T (Reactor Water Cleanup System Pipe Breaks) and a harsh radiation environment following a PBIC and all other PBOC's.

From FSAR Section 5, the Containment Spray Subsystem provides containment spray capability as an alternate method for reducing containment pressure following a LOCA. This subsystem is designed to remove energy from the drywell by condensing steam or to cool noncondensable gases which have collected in the suppression pool. Since a PBOC does not result in these conditions, these pressure switches do not need to be able to withstand the environmental conditions associated with PBOC's.

For PBIC's, radiation levels of 1×10^6 rads are not reached until at least 100 hours after the pipe break occurs. As per GE letter the limiting materials are neoprene and Buna-N. Per D.O.R. Guidelines, Buna-N has a radiation threshold of 1×10^6 rads and neoprene a radiation threshold of 1×10^7 rads. Therefore, switch failure due to radiation would not be expected to occur until at least 100 hours after the pipe break. It can reasonably be assumed that the containment spray subsystem would not be required at this time.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1360-9 (A-D)
TER No. 194 Sheet 1 of 3

Preparer: JLRogers Date: 7-30-84
Independent Review: MR Ewin Date: 7/30/84
Approval: RCDonnay Date: 7/30/84

PS1360-9 (A-D) are Barksdale model P1H-M85SS-V pressure switches used to sense low pressure in the steam line supply lines to the RCIC pump turbine. The switches are used to signal the closure of two motor operated valves in the RCIC steam supply line in order to prevent steam and radioactive gases from escaping through the turbine shaft seals into the reactor building over a 30 day mission length. This protection is only required after reactor steam pressure has decreased to such a low value that the turbine can no longer be operated (approximately 100 psig or less). This condition is expected to be reached during reactor vessel cooldown and depressurization within a few minutes following a LOCA or approximately 6-8 hrs following a small break PBIC or PBOC. It is expected that the reactor vessel will reach atmospheric pressure approximately 2-4 hours later at which time, this protective function will no longer be required. These switches are mounted in the RCIC steam leak instrument rack (C2257-B) located in the mezzanine of the RCIC quadrant. These switches could be exposed to a harsh superheated steam and radiation environment during a PBOC-5 (HPCI steam line break in the torus compartment) or PBOC-6 (RCIC steam line leak in the RCIC pump room) or to solely a harsh radiation environment during any other PBOC or a PBIC. Continued operation with these switches can be justified based on the following analyses.

Justification

o Temperature and Humidity

The PBOC-5 service profile (246°F maximum and 100% RH) exceeds the test profile (extended exposure to saturated steam at 212°F) for approximately the first 3 minutes of the transient. However, the thermal inertia of the switch and instrument rack in the presence of superheated steam should result in the temperatures actually experienced in the vital portions of the switch being enveloped by the test profile. In the unlikely event the switches did fail, two scenarios could occur. If the switch failed closed, RCIC, which is not credited for this transient, would remain isolated. If the switch failed open, the control room operator could be reasonably assumed to close the valves several hours later following reactor vessel cooldown/depressurization and termination of RCIC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1360-9 (A-D)
TER No. 194 Sheet 2 of 3

Preparer:	JL Rogers	Date:	7-30-84
Independent Review:	NR Ervin	Date:	7/30/84
Approval:	RC Denny	Date:	7/30/84

The PBOC-6 service profile (short term exposure to 155°F and 90% humidity) is less severe than the test profile (extended exposure at 212°F and 100% relative humidity) as documented in AETL Test Report 596-0398. Therefore, the test temperature profile in the test is actually more severe than the service condition and continued operation of the plant is justified.

o Pressure

The service profile reaches a peak of 14.9 psia, whereas the test pressure reaches a maximum of 7" H_2O (14.95 psia). Based on this fact and due to weathertight construction of this instrument, in our engineering judgment, no functional disparities will occur. Therefore, continued plant operation is justified.

o Radiation

From a Wyle Labs analysis of the materials used for construction of this component, the most limiting material is a fiberboard type insulator which has a damage threshold of approximately 10^6 rads for this application. The leak tight nature of the switch is expected to preclude beta radiation exposure to this components. It is estimated that the gamma dose to this component will meet the 10^6 threshold approximately 400 hours following a LOCA. It is expected that the RCIC steam line would have isolated on low pressure and sealed in prior to this time. In the event that this exposure induced a failure of the switch, the steam line valves would be capable of opening if the operator reset the isolation and either the operator held the control switches in the open position or a low reactor vessel level or high drywell pressure was sensed. In the unlikely event that these conditions were met, opening of the valves would have a negligible effect on the ability to maintain exposures below 10CFR100 limits since RCIC operation should be completed and the reactor vessel would be expected to be nearing a cold depressurized condition at the time the damage threshold would be reached. In addition, failure of the switch would not inhibit the ability to reclose these valves. Continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1360-9 (A-D)
TER No. 194 Sheet 3 of 3

Preparer:	JcRogers	Date:	7-30-84
Independent Review:	NR Ern	Date:	7/30/84
Approval:	RC Denny	Date:	7/30/84

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o Aging

Wyle Labs has completed aging and thermal degradation analyses that confirm that the six hour qualification exposure documented in AETL Test Report 596-0398, envelopes the conditions experienced at PNPS over 40 years and a 30 day mission length accident exposure for these switches.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS2360-1
TER No. 196 Sheet 1 of 2

Preparer: N R Ers Date: 7/26/84

Independent Review: WA Clancy Date: 7/26/84

Approval: Date: 7/27/84

This pressure switch detects HPCI pump low suction pressure and is, therefore, required solely to assure satisfactory HPCIS operation.

The HPCIS is relied upon to operate during and following

Loss of Feedwater Flow,
Total Loss of Offsite Power,
Shutdown from Outside Control Room (Special Event),
Pipe Break Inside Primary Containment,
Control Rod Drop Accident, and
Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of this pressure switch is the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS2360-1
TER No. 196 Sheet 2 of 2

Preparer: NR Ervin Date: 7/26/84

Independent Review: W S Clancy Date: 7/26/84

Approval: RC Denny Date: 7/27/84

On the other hand, system operability is required for the main steam line breaks, PBOC-7 and PBOC-8, either of which could result in cumulative radiation exposures to the pressure switch well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of PS2360-1, as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS-2389 (A-D)
TER No. 207 Sheet 1 of 2

Preparer:	7/R Ervin	Date:	8/2/84
Independent Review:	JL Rogers	Date:	8/2/84
Approval:	RC Denny	Date:	8/2/84

PS-2389(A-D) are Barksdale model P1H-M85SS-V pressure switches used to sense low pressure in the steam line supply lines to the HPCI pump turbine. The switches are used to signal the closure of two motor operated valves in the HPCI steam supply line in order to prevent steam and radioactive gases from escaping through the turbine shaft seals into the reactor building. This protection is only required after reactor steam pressure has decreased to such a low value that the turbine can no longer be operated (approximately 100 psig or less). This condition is expected to be reached during reactor vessel cooldown and depressurization within a few minutes following a LOCA or approximately 6-8 hrs following a small break PBIC or PBOC. It is expected that the reactor vessel will reach atmospheric pressure approximately 2-4 hours later at which time, this protective function will no longer be required. These switches are mounted in the HPCI steam leak instrument rack (C2257-A) located in the NW RHR quadrant. These switches could be exposed to a harsh superheated steam and radiation environment during a PBOC-5 (HPCI steam line break in the torus compartment) or to solely a harsh radiation environment during any other PBOC or a PBIC. Continued operation with these switches can be justified based on the following analyses.

Analytical Justification

o Temperature and Humidity

AETL Test Report 596-0398 documents the qualification testing of a similar component in a harsh steam environment. The test profile consisted of a rise to 212°F in an unspecified time with a dwell at 212°F for six hours. However, the PBOC-5 Service profile for this location consists of a rapid spike to 229°F with a return to less than 212°F in less than 3 minutes. It is our engineering judgment that due to the thermal inertia of the components, the internal temperature experienced by these switches during the predicted service event will be significantly less than that which was experienced in the test. Therefore, the test temperature profile is essentially more severe than the service conditions and continued operation of the plant is justified. It should be noted that HPCI operation is not required during this transient and that the valves controlled by these switches will be automatically closed in response to increased HPCI flow and space temperature resulting from the leak.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS-2389 (A-D)
TER No. 207 Sheet 2 of 2

Preparer:	NR Ervin	Date:	8/2/84
Independent Review:	JL Rogers	Date:	8/2/84
Approval:	RCMcDenny	Date:	8/2/84

o Pressure

The service profile reaches a peak pressure of 15.3 psia and decays to atmospheric pressure within seconds. In our engineering judgment, exposure to this pressure change will cause no functional disparities due to weathertight construction of these switches. It should be noted that HPCI operation is not required during this transient and that the valves controlled by these switches will be automatically closed in response to increased HPCI flow and space temperature resulting from the leak. Therefore, continued plant operation is justified.

o Radiation

From a Wyle Labs analysis of the materials used for construction of this component, the most limiting material is a fiberboard type insulator which has a damage threshold of approximately 10^6 rads for this application. The leak tight nature of the switch is expected to preclude beta radiation exposure to this components. It is estimated that the gamma dose to this component will meet the 10^6 threshold approximately 5 hours following a LOCA. It is expected that the HPCI steam line would have isolated on low pressure prior to this time. In the unlikely event that this exposure induced a failure of the switch, the steam line valves would be capable of opening if the operator held the control switches in the open position or if a low reactor vessel level or high drywell pressure is sensed. In the unlikely event that these conditions were met, opening of the valves would have a negligible effect on the ability to maintain exposures below 10CFR100 limits since HPCI operation should be completed and the reactor vessel would be expected to be nearing a cold depressurized condition at the time the damage threshold would be reached. In addition, failure of the switch would not inhibit the operators ability to reclose these valves. Continued operation is justified.

o Aging

Wyle Labs has completed aging and thermal degradation analyses that confirm that the six hour qualification exposure documented in AETL Test Report 596-0398, envelopes the conditions experienced at PNPS over 40 years and a 30 day mission length accident exposure for these switches.

BOSTON EDISON COMPANY
JUSTIFICATION FOR CONTINUED OPERATION

Equipment Identification No. M01301-62
TER No. N/A

Preparer: JL Rogers Date: 7/30/84

Independent Review: NR Ein Date: 7/30/84

Approval: RC Denny Date: 7/30/84

M01301-62 operates the block/control valve for supplying cooling water flow from the RCIC Pump Discharge to the Barometric Condenser (E-201). The valve is normally closed but must open to facilitate RCIC operation. The valve is located in the RCIC Pump Room. The only transients for which RCIC operation is credited are control rod drop, loss of all AC power and loss of feedwater.

However, these transients have been evaluated and no RCICs equipment will be subject to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation. Therefore operability of M01301-62 during exposure to a harsh environment need not be demonstrated for transients requiring RCIC operation.

Therefore, this valve is not within the scope of 10CFR50.49 and will be deleted from the EQML.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01301-49
TER No. N/A Sheet 1 of 1

Preparer: JE Rogers Date: 7/30/84

Independent Review: NR Eis Date: 7/30/84

Approval: [signature] Date: 7/30/84

M01301-49 operates the block valve in the discharge of the RCIC pump. This valve is located outside the containment in the RCIC Pump Room (zone 1.10). M01301-49 utilizes a 125v DC reliance motor with class "HR" insulation for which complete qualification documentation is not available. M01301-49 is normally closed and is automatically signaled opened in response to a low reactor vessel level to facilitate injection of RCIC coolant into the vessel. The only transients for which RCIC operation is credited are Control Rod Drop, total loss of AC power and loss of feedwater.

However, these transients have been evaluated and no RCICs equipment will be subject to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation. Therefore operability of M01301-49 during exposure to a harsh environment need not be demonstrated for transients requiring RCIC operation.

M01301-49 serves a second safety related function by providing containment isolation for transients not requiring RCIC operation. M01301-49 would remain in a normally closed position during such transients and would not be required to actively function. In the unlikely event that M01301-49 was open and failed open, redundant isolation of this penetration would be provided by M01301-48, A01301-50 and 58A.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-19
TER No. N/A Sheet 1 of 1

Preparer: JL Rogers Date: 7-30-84

Independent Review: NR Ers... Date: 7/30/84

Approval: RCM... Date: 7/30/84

MO1001-19 operates the block valve in the cross tie between the "A" and "B" train RHR pump combined discharge headers. The valve is located in the CRD mezzanine pump room at elevation 2'9" (zone 1.8). The valve is normally key-locked open and should remain open to ensure that all four RHR pumps deliver LPCI to the selected loop. This valve could be exposed to a harsh steam and radiation environment during a PBOC-5 (HPCI Steam Line Break in the Torus Compartment) and/or to a harsh radiation environment during a PBIC, or any PBOC other than a PBOC-5. The valve is required to remain functional for a 30 day mission time to facilitate operation of the RHR system in a variety of modes. Limitorque Report B0003 documents the qualification of a similar operator and motor for a harsh steam and radiation environment that envelopes the service profile for all postulated transients affecting MO1001-19 including the PBOC-5. MO1001-19 is therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were utilized for power lead termination (required by IE Notice 83-72). Inspection of the terminal blocks in the Limitorque operators is in progress. No deficiencies in the terminal blocks have been reported and therefore there is a high degree of assurance that the terminal blocks in this motor operator are qualified. Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-6
TER No. N/A   Sheet 1 of 1

Preparer:	W S Cleary	Date:	20 JUNE 84
Independent Review:	JL Rogers	Date:	20 JUNE 84
Approval:	RrGrayno	Date:	6/23/84

MO2301-6 operates the valve in the line from the condensate storage tanks to the HPCI pump suction. The valve is located in the RBCCW pump room-B (zone 1.22) and is normally open. The valve will automatically be closed to facilitate a transfer of HPCI suction to the torus in the event that high torus level or low condensate storage tank level is sensed.

During a PBOC-3 (HPCI Steam Line Break in the HPCI Pump Station) this valve would be exposed to a harsh environment. However, HPCI operation is not required for this transient so the exposure of this valve is inconsequential since it serves no function other than supporting HPCI operation.

During any other PBOC, a PBIC or a Control Rod Drop, this valve must remain operable over the five (5) hours mission time of the HPCI System. However, analysis has indicated that the valve would not be exposed to a harsh environment during this time frame.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO4065
TER No. N/A
Sheet 1 of 1

Preparer: JLRogers Date: 7-30-84

Independent Review: NR Ersin Date: 7/30/84

Approval: RADenny Date: 7/30/84

MO4065 operates the block valve in the RBCCW supply line to fuel pool heat exchanger E-206A. This valve is normally open and needs to be capable of closing to reduce non-essential loads on the RBCCW system during a design basis event requiring RBCCW cooling water supply to the RHR heat exchanger(s). This valve is located outside containment in the fuel pool heat exchanger room (zone 1.13).

A valve operator and motor similar to MO4065 was qualified in a steam environment to 250°F, 25 psig and 2×10^7 rads as documented in Limitorque Report B0003. This qualification profile envelopes the service profile for all potential harsh environment exposures to MO4065 and is therefore considered qualified pending completion of an inspection to verify that appropriate terminal blocks were utilized for power lead termination (required by IE Notice 83-72). Inspection of the terminal blocks in the Limitorque operators is in progress. No deficiencies in the terminal blocks have been reported and therefore there is a high degree of assurance that the terminal blocks in this motor operator are qualified. Based on this determination, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO4009A, MO4009B
TER No. N/A   Sheet 1 of 1

Preparer: JLRogers   Date: 8/1/84

Independent Review: NR Ev   Date: 8/1/84

Approval: RCDerry   Date: 8/2/84

MO4009A and MO4009B are required for isolation of non-essential loads in the "B" Loop of the RBCCW System and may also be used for subsequent restoration of non-essential RBCCW loads once the heat load of the RHR heat exchangers has decreased. These valves are located outside containment in the RBCCW Pump Rocm-B and are normally open.

During a PBIC, these valves will be exposed to increased amounts of radiation. However, the increase will be insufficient to cause a harsh environment exposure until after the valves mission length has elapsed. As a result, the operability of MO4009A/B post-PBIC is assured.

A valve operator and motor combination similar to MO4009A and MO4009B was qualified for extended exposure to steam at 250°F as documented in Limitorque Qualification Test Report B0003. During a PBOC-3, MO4009A and MO4009B are exposed to superheated steam resulting in a service profile which peaks at 301°F and exceeds the qualification profile for approximately the first five minutes. However, the short term thermal inertia of the valve operator in a superheated steam environment, as documented in Limitorque Report B0027, would result in the vital portions of the operator and motor being exposed to temperatures well below those of the qualification profile. In addition, the valves would not be exposed to a harsh radiation environment. Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO4083
TER No. N/A Sheet 1 of 1

Preparer: JL Rogers Date: 8/1/84

Independent Review: NR Eisen Date: 8/1/84

Approval: RCDenny Date: 8/1/84

MO4083 operates the RBCCW bypass valve around the "B" loop RBCCW to Salt Service Water Heat Exchanger E-209B. The valve is located in the RBCCW-B compartment and is normally maintained in a throttled position. If the valve failed to operate, the ability of the RHR system to cool the torus would be degraded.

During a PBIC, this valve is exposed to increased radiation. However, a harsh exposure does not occur until well after the 30 day mission life of these valves has passed. As a result, operability of this valve post-PBIC is assured.

A valve operator and motor similar to MO4083 was qualified for extended exposure to saturated steam at 250°F as documented in Limitorque Qualification Test Report B0003. During a PBOC-3, MO4083 is exposed to superheated steam resulting in a service profile which peaks at 301°F and exceeds the qualification test profile for approximately the first 5 minutes. However, the short term thermal inertia of the valve operator in a superheated steam environment as documented in Limitorque Report B0027, would result in the vital portions of the operator and motor being exposed to temperatures well below those of the qualification test profile. In addition, the valve would not be exposed to a harsh radiation environment during the 30 day mission length. Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO3805, MO3806
TER No. N/A   Sheet 1 of 1

Preparer:	WA Chang	Date:	7/27/84
Independent Review:	NR Ein	Date:	7/27/84
Approval:	RC Denny	Date:	7/30/84

MO3805 and MO3806 operate the block/isolation valves in the salt service water discharge from TBCCW heat exchanger E-122B and RBCCW heat exchanger E-209B respectively. These valves are located in the RBCCW Pump Room-B and are normally maintained in a throttled open position. Following a design basis accident, MO3805 will close to a preset throttle position and MO3806 will go full open to ensure an adequate supply of salt service water to the RBCCW heat exchanger to facilitate LPCI, Core Spray or RHR torus cooling operation as required. Both operators are equipped with Reliance Motors Utilizing Class B Insulation for outside containment application.

During a PBIC, these valves are exposed to increased radiation. However, a harsh radiation environment does not develop until well after the 30 day mission life of these valves has passed and as a result, MO3805 and MO3806 need not be qualified for a PBIC.

A valve operator and motor combination similar to MO3805 and MO3806 was qualified for extended exposure to steam at 250°F as documented in Limitorque Report B0003. During a PBOC-3, MO3805 and 3806 are exposed to superheated steam resulting in a service profile which peaks at 301°F and exceeds the qualification profile for approximately the first five minutes. However, the short term thermal inertia of the valve operator in a superheated steam environment, as documented in Limitorque Report B0027, would result in the vital portions of the operator and motor being exposed to temperatures well below those of the qualification profile. In addition, the valves would not be exposed to a harsh radiation environment. Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. CX2, CX4, 520, CX8
TER No. N/A
Sheet 1 of 1

Preparer: NR Ervin Date: 7/27/84

Independent Review: WS Clancy Date: 7/27/84

Approval: RA Denny Date: 7/27/84

This equipment consists of polyethylene insulated cable (installed outside of the drywell) provided by several manufacturers. While no qualification documentation or testing history has been found for these specific cables, similarly constructed cable has been successfully subjected to sequential testing (proprietary TR #17513-1), which documents qualification of the insulation system to 1.63 x 10^8 rads gamma and a LOCA condition including temperatures up to 325°F. These conditions are more severe than the conditions at PNPS.

The generic materials which make up the insulation system have expected lives of greater than 1.5E4 years (PE) in an ambient temperature of 105°F.

Based on the above, it is judged that the PE cable installed is justified for continued use pending further testing or replacement.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. S1, S37, CXG, Z3, Z3A, S48, C02, C03, S19, S27
TER No. N/A Sheet 1 of 1

Preparer: TR Ein Date: 8/1/84

Independent Review: JL Rogers Date: 8/1/84

Approval: RE Denny Date: 8/1/84

This equipment consists of polyethylene insulated polyvinylchloride jacketed cable (installed outside the drywell) provided by several manufacturers. While no qualification documentation or testing history has been found for these specific cables, similarly constructed cable has been successfully subjected to sequential testing (proprietary TR #17513-1), which documents qualification of the insulation system to 1.63×10^8 rads gamma and a LOCA condition including temperatures up to 325°F. These conditions are more severe than the conditions at PNPS.

The generic materials which make up the insulation system have expected lives of greater than 1.4E4 years (PVC) and greater than 1.5E4 years (PE) in an ambient temperature of 105°F.

Based on the above, it is judged that the PVC/PE cable installed is justified for continued use pending further testing or replacement.

BOSTON EDISON COMPANY
JUSTIFICATION FOR CONTINUED OPERATION

Equipment Identification No. SVL61
TER No. N/A   Sheet 1 of 1

Preparer: NR Ei[illegible]   Date: 7/30/84

Independent Review: JL Rogers   Date: 7/30/84

Approval: [illegible]   Date: 7/30/84

This solenoid valve is an ASCO HT8320A10T. It is unknown if this valve will survive the radiation due to a LOCA; however, it is not required to operate post- accident. SVL61 controls the air supply to operate the cross-tie damper between Reactor Building Clean Exhaust and Refueling Floor Exhaust Ducts. This damper provides the isolation between safety-related and nonsafety-related exhaust ducts. If the SVL61 disc fails, the air supply to AO/N-138 will be vented and AO/N-138, which is normally closed, will fail open. If AO/N-138 fails open, when SGTS operates, a suction will be drawn on both the Reactor Building Clean Exhaust and the Refueling Floor Exhaust simultaneously.

If SVL61 fails such that air is continuously supplied to AO/N-138, AO/N-138 will remain shut. However, the Reactor Building Clean Exhaust and Refueling Floor Exhaust can both be ventilated through their own ducts and isolation valves (AO/N-100 and AO/N-101).

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. SVL82, SVL83
TER No. N/A Sheet 1 of 1

Preparer: N.R. E[illegible] Date: 6/6/84

Independent Review: W S[illegible] Date: 6/6/84

Approval: R[illegible] Date: 7/26/84

These solenoid valves control air operated valves that allow the SGTS to obtain a suction on the suppression pool. These valves are not required to operate post-PBOC; therefore, they only need to be qualified for radiation. The 40 year TID plus LOCA dose is 2.86×10^6 rads. The two materials of concern in this valve are the Buna-N elastomers and the acetal disc holder. In EPRI Report NP-2129, "Radiation Effects on Organic Materials in Nuclear Plants" Buna-N is listed as not reaching the 50% loss of elongation point until after 7×10^7 rads and acetal resins are listed as reaching the 50% loss of tensile strength at 4×10^6 rads. Each of these doses is greater than the combined 40 year TID plus accident dose. Therefore, no detrimental effects, due to radiation, are expected, and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. P202D, E, F
TER No. N/A Sheet 1 of 1

Preparer: T/R Ers— Date: 7/27/84

Independent Review: WA Clumy Date: 7/27/84

Approval: RADenny Date: 7/30/84

These pumps/motors provide the motive force for the RBCCW flow in RBCCW Loop 'B'. P202E and F have motors that are environmentally qualified. P202D has a motor that has been rewound. All three motors have motor lead connections that have not been environmentally qualified. However, a justification for continued operation has been previously presented (Terminations-Ring Tongue (<4KV)) for the connections.

Each RBCCW loop is designed to transfer the RHR system heat load (64×10^6 BTUs/hour) plus an additional heat load of 1×10^6 BTUs/hour for other essential equipment with only 2 of the 3 RBCCW pumps in the loop operating. Therefore, if P202D did fail post-accident, the operators would be able to start the non-operating pump in the 'B' Loop and /or shift to the 'A' Loop. Based on the above information, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. 412A Okonite, 106 Kerite, 212B Rockbestos, SC16 Rockbestos
TER No. N/A Sheet 1 of 1

Preparer: MR Ern Date: 7/27/84

Independent Review: WA Clancy Date: 7/27/84

Approval: RCDenny Date: 7/27/84

The insulation systems used on these cables are environmentally qualified. The documentation packages for these cables are being completed and when completed, environmental qualification will be proven. Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J720
TER No. N/A Sheet 1 of 1

Preparer: NR Ervin Date: 8/1/84

Independent Review: JL Rogers Date: 8/1/84

Approval: [signature] Date: 8/1/84

This junction box is in the electrical circuit on MO/N-113 and SVL-70. MO/N-113 and SVL-70 are only required post-LOCA. Therefore, this junction box need only be qualified for radiation.

The manufacturer of terminal blocks installed in this junction box has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J32
TER No. N/A Sheet 1 of 1

Preparer: Norm R Ers Date: 6/29/84

Independent Review: W A Cheng Date: 7/19/84

Approval: RC Denny Date: 7/26/84

This junction box is in the electrical circuit of the following temperature switches:

Switch	Location	Setpoint
TS2370C	HPCI Valve Room exh. duct	160°F-170°F
TS2371C	Torus Area exh. duct	190°F-200°F
TS2372C	HPCI Valve Room exh. duct	160°F-170°F
TS2373C	Torus Area exh. duct	190°F-200°F

Exceeding the setpoint is an indication of a HPCI steam line break. The closing of the temperature switch contacts causes an auto isolation signal (which "seals in") to be sent to MO2301-4, 5, 35, and 36, which shut to isolate the HPCI steam supply line and the HPCI pump suction line from the Torus. The terminal block inside the junction box need only operate up to the point that the isolation signal "seals in". Because the terminal block is inside the junction box there will be a time lag between the temperature switches being subjected to the pipe break and the terminal blocks being subjected to the pipe break. The terminal blocks will probably never be subjected to an elevated temperature prior to performing their safety function. Therefore, failure of the terminal block is unlikely and continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J217
TER No. N/A
Sheet 1 of 1

Preparer: NR Ervin Date: 7/27/84
Independent Review: WA Clancy Date: 7/27/84
Approval: [signature] Date: 7/27/84

This junction box is in the electrical circuit for solenoid valve SV220-45, that in turn controls AO220-45 (Reactor Coolant Sample Line Outboard Isolation Valve). SV220-45 is normally shut and is required to be shut to ensure primary containment isolation post-accident. If AO220-45 is open at the beginning of an accident, the isolation signal causes SV220-45 to deenergize which will cause AO220-45 to shut. All credible failures of the terminal block within J217 will also deenergize SV220-45 and cause AO220-45 to shut. Therefore, failure of J217 will have no adverse effects, and continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J451
TER No. N/A Sheet 1 of 1

Preparer: Norm R. Ervin Date: 7/24/84

Independent Review: WS Clancy Date: 7/24/84

Approval: RC Denny Date: 7/27/84

This junction box is in the electrical circuit for the following valves:

SV5033B	Normal Purge Suppy to Drywell
SV5033C	Normal Nitrogen Makeup to Drywell
SV5035B	Purge Air to Drywell
SV5036B	Purge Air to Torus
SV5042B	Purge Exhaust from Torus
SV5044B	Purge Exhaust from Drywell

SV5033C is normally energized and AO5033C is normally open. All other valves are normally deenergized and their respective air operated valves are normally shut. All of the air operators fail shut. All of the solenoid valves receive Containment Isolation signals upon a LOCA. The isolation signal causes the solenoid valves to deenergize (if not already deenergized) and the air operated valves to shut. All credible failures of the terminal block within J451 will also deenergize the solenoid valves and cause the air operated valve to shut. Therefore, failure of J451 will have no adverse effects, and continued plant operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J866
TER No. N/A Sheet 1 of 1

Preparer: NR Ervin Date: 7/30/84

Independent Review: JE Rogers Date: 7/30/84

Approval: [signature] Date: 7/30/84

The manufacturer of the terminal block in this junction box has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 189.6°F and immediately starts to decrease. Applying 189.6°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is less than .00075, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreases to 140°F within 60 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C2257B
TER No. N/A — Sheet 1 of 1

Preparer: TR Ev — Date: 7/19/84

Independent Review: W S Clancy — Date: 7/19/84

Approval: [signature] — Date: 7/26/84

The manufacturer of terminal blocks in this local control panel has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 246.5°F and immediately starts to decrease. Applying 246.5°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is only .0018, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreased to 122°F within 60 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C2207A
TER No. N/A

Sheet 1 of 1

Preparer: NR Ex— Date: 7/19/84

Independent Review: WS Cary Date: 7/19/84

Approval: RC Denny Date: 7/26/84

The manufacturer of terminal blocks in this local control panel has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 246.3°F and immediately starts to decrease. Applying 246.3°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is only .0018, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreases to 123°F within 60 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J599, J600, J601, J602, J606
TER No. N/A

Preparer: [signature] Date: 7/17/84

Independent Review: W.S. Clancy Date: 7/19/84

Approval: [signature] Date: 7/26/84

The manufacturer of terminal blocks installed in these junction boxes has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 224.7°F and immediately starts to decrease. Applying 224.7°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is only .0018, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreased to 132°F within 90 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS4008
TER No. N/A

Sheet 1 of 1

Preparer: NR Ervin Date: 7/27/84

Independent Review: WS Clancy Date: 7/27/84

Approval: RC Denny Date: 7/27/84

This pressure switch sends a pump low discharge pressure signal for pumps P202D, E, F to the RBCCW Loop 'B' Control Circuitry. When the discharge pressure goes below 31 psig the pressure switch sends a signal to start either pump P202E or F, whichever is in "auto". Failure of the pressure switch may prevent the pumps from starting automatically upon failure of one pump. However, each RBCCW loop is designed to transfer the RHR system heat load (64 x 10^6 BTUs/hour) plus an additional heat load of 1 x 10^6 BTU/hour for other essential equipment. Therefore, if failure of the pressure switch, along with a PBOC and failure of the operating RBCCW Loop 'B' pump does occur, the operator can shift to RBCCW Loop 'A' to remove the heat loads. The Loop 'A' equipment is located in an area of mild environment and therefore can be relied on to operate post-accident. Based on the above information, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. TIP Ball and Shear Valves
TER No. N/A Sheet 1 of 1

Preparer:	Thomas Frisardi	Date:	July 27, 1984
Independent Review:	W S Clancy	Date:	7/27/84
Approval:	R C Denny	Date:	7/30/84

The safety function of these valves is primary containment isolation. The only accident for which they must provide this safety function is a pipe break inside primary containment. The components are located outside containment, and therefore, they must be qualified for radiation and aging only.

Continued operation with these components not qualified is justified because they provide diverse means of isolating the affected penetrations. The ball valves are closed more than 99% of the time (TIP usage is approximately 3 hrs. per 2 week period) and they do not require power to remain closed. In the unlikely event that a pipe break inside primary containment occurs with the TIP probes inside the drywell, diversity in the system provides assurance that the penetrations will be isolated. The limit switches in the ball valve provide a signal to the Primary Containment Isolation display in the Main Control Room. Should any of the ball valves spuriously open or be held open by a stuck probe under accident conditions, the operators can detect this and fire the shear valves which are powered by 125V DC, ensuring operability in case offsite power is lost.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J444, J463
TER No. N/A
Sheet 1 of 1

Preparer:	NR Ervin	Date:	8/1/84
Independent Review:	JL Rogers	Date:	8/1/84
Approval:	RC Denny	Date:	8/1/84

These junction boxes are in the electrical circuits for solenoid valves that are required to deenergize to shut Reactor Building isolation dampers. The dampers shut immediately after low reactor water level, high drywell pressure or high radiation in refueling floor exhaust duct is sensed. If the terminal block inside the junction box fails after the dampers are shut, there will be no detrimental effects because the dampers are already shut. In the unlikely event that the terminal block fails prior to the solenoids deenergizing and the dampers shutting, the failure will simply speed up the process of isolating the Reactor Building. After the dampers have shut, there is no plausible failure of the terminal block that could reopen the valve. Therefore, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J561, J604, J874, J603, J553
TER No. N/A Sheet 1 of 1

Preparer:	NR Ein	Date:	8/1/84
Independent Review:	JL Rogers	Date:	8/1/84
Approval:	RA Kenny	Date:	8/1/84

The manufacturer of terminal blocks installed in these junction boxes has not yet been determined. However, Sandia National Laboratories and other labcratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 238.1°F and immediately starts to decrease. Applying 238.1°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is only .0018, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreased to 132°F within 90 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure. Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J859
TER No. N/A   Sheet 1 of 1

Preparer: NR Ervin   Date: 8/1/84

Independent Review: JLRogers   Date: 8/1/84

Approval: [signature]   Date: 8/1/84

The insulation system used on the splices in this junction box are environmentally qualified. The documentation packages for these splices are being completed and when completed, environmental qualification of the splices and this junction box will be proven. Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. N912, N923
TER No. N/A
Sheet 1 of 2

Preparer: TR Ee___ Date: 7/27/84

Independent Review: WS Clancy Date: 7/27/84

Approval: _________ Date: 7/30/84

N912 is a local start switch for P223 (Gland Seal Condenser Blower) and N923 is a local start switch for P220 (Gland Seal Condenser Condensate Pump).

These switches contribute to HPCIS turbine operation and are, therefore, required solely to assure satisfactory HPCIS operation.

The HPCIS is relied upon to operate during and following

Loss of Feedwater Flow,
Total Loss of Offsite Power,
Shutdown from Outside Control Room (Special Event),
Pipe Break Inside Primary Containment,
Control Rod Drop Accident, and
Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of these switches are the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. N912, N923
TER No. N/A

Preparer:	?/R E[illegible]	Date:	7/27/84
Independent Review:	WA Clancy	Date:	7/27/84
Approval:	RC[illegible]	Date:	7/30/84

On the other hand, system operability is required for the main steam line breaks, PBOC-7 and PBOC-8, either of which could result in cumulative radiation exposures to the switches well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of N912 and N923, as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. N921
TER No. N/A

Preparer: NR Ers[illegible] Date: 7/30/84

Independent Review: JL Rogers Date: 7/30/84

Approval: [illegible] Date: 7/30/84

N921 is a local switch for P229 (HPCI Turbine Auxiliary Oil Pump). P229 is only required during turbine startup and shutdown.

The HPCIS is relied upon to operate during and following

Loss of Feedwater Flow,
Total Loss of Offsite Power,
Shutdown from Outside Control Room (Special Event),
Pipe Break Inside Primary Containment,
Control Rod Drop Accident, and
Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of this switch is the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. N921
TEP No. N/A

Preparer: NR Ervin Date: 7/30/84

Independent Review: JE Rogers Date: 7/30/84

Approval: RC Denny Date: 7/30/84

On the other hand, system operability is required for the main steam line breaks, PBOC-7 and PBOC-8, either of which could result in cumulative radiation exposures to the switch well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of N921, as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. CS42-1821, CS42-1822
TER No. N/A Sheet 1 of 1

Preparer: NR Ervin Date: 7/24/84

Independent Review: W S Cleary Date: 7/24/84

Approval: RC Denny Date: 7/26/84

The functional requirement of these switches (that control VAC201) is that normally closed contacts internal to the switches remain shut. The switches are mounted in an enclosed control panel. The non-metallic portion of the switch is made of Dupont Delrin.

The only way the contacts could open would be for catastrophic failure of the Delrin. The parameters that could cause catastrophic failure, would be temperature (Delrin softening or embrittling) or radiation (Delrin disintegrating). The radiation to which the switch might be subjected is 1.6×10^5 rads, but it has been tested to 1×10^6 rads, therefore radiation is not a problem. The temperature due to the worst case postulated break is 238.1°F, 24.5 seconds into the accident, and considering that Delrin has been tested to a much higher temperature (311°F) temperature is not a problem. Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C61A, C61B
Indicating Lights
TER No. N/A
Sheet 1 of 1

Preparer: MR Erwin Date: 7/30/84

Independent Review: JL Rogers Date: 7/30/84

Approval: [signature] Date: 7/30/84

These lights are for indication at the local control panels. They are not necessary for operation of the fans, however, failure could affect the control circuit and therefore degrade operation of their respective area cooling fans. There are three possible failure modes; open, short internal to the light, or short to the panel.

If a light fails open there will be no effect on the circuit and the fan will continue to operate normally.

If a light develops an internal short circuit there will be no effect on the circuit and the fan will continue to operate normally.

If a light develops a short to the panel the control circuit will be disabled and the fan will be deenergized. However, the only possible failure mechanism would be excessive moisture inside the panel such that the water created a path for current from the light connections to the panel. This failure mechanism is extremely unlikely because the panels are gasketed, which will reduce the moisture inside, and because of the distance between the electrical connections and the panel. The distance is enough so that creation of an electrical arc between the connections and the panel is extremely small. Also, if one of the fans becomes disabled it is highly unlikely that this would cause a significant effect on the equipment in the affected room, because the RHR rooms (areas 1.1 and 1.2) have redundant fans.

Based on the above information, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C2257A

TER No. N/A Sheet 1 of 1

Preparer: Norman R Ev[illegible] Date: 7/19/84

Independent Review: W S Clancy Date: 7/19/84

Approval: R C McKenny Date: 7/26/84

The manufacturer of one of the terminal blocks within this local control panel has not yet been determined. However, Sandia National Laboratories and other laboratories have compiled extensive test data on terminal blocks (both protected and unprotected) of various manufacturers, which has shown that the probability of failure of the terminal blocks is very low. Sandia tested over 400 terminals in their own facilities.

The worst case temperature peaks at 228.7°F and immediately starts to decrease. Applying 228.7°F to Sandia's probability curves for 480v/100%RH/5 hours, the probability of failure is only .0018, for protected terminal blocks in 120V circuits. The installed terminal blocks may be required for longer than 5 hours, but the Reactor Building temperature decreased to 119°F within 60 minutes of the PBOC, thereby reducing the probability of failure even further. Also, Sandia's research showed that most failures occurred in the initial phase of the tests.

Sandia found "that for TMI-2 accident conditions, the effect of radiolysis is negligible on surface conductivity and therefore on breakdown" and therefore the radiation in the Reactor Building will not increase the probability of failure.

The manufacturer of the instrument rack wire from the terminal blocks AA and DD to DPIS2353 and DPIS2352, respectively, is unknown. The wire is in a conduit from the switches to the enclosure for the terminal block. The differential pressure switches are only required to provide a signal (which "seals in") to the Primary Containment Isolation Control System during a Main Steam Line Break in the Steam Tunnel (PBOC-8). The only environmental changes in the location of C2257A during the 1 minute that the pressure switches are required, are a pressure spike to 15.2 psia and a temperature of 115.9°F. These "mild" conditions will not add any abnormal stresses to the wire and the wire is expected to survive the accident.

Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. P229
TER No. N/A

Preparer:	MR Ers[illegible]	Date:	7/27/84
Independent Review:	WA Clancy	Date:	7/27/84
Approval:	RC McCurry	Date:	7/30/84

P229 is the HPCI Turbine Auxiliary Oil Pump. It is only required during turbine startup and shutdown.

The HPCIS is relied upon to operate during and following

Loss of Feedwater Flow,
Total Loss of Offsite Power,
Shutdown from Outside Control Room (Special Event),
Pipe Break Inside Primary Containment,
Control Rod Drop Accident, and
Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of this pump is the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. P229
TER No. N/A   Sheet 2 of 2

Preparer: NR Ersi   Date: 7/27/84

Independent Review: WL Clancy   Date: 7/27/84

Approval: RC Denny   Date: 7/30/84

On the other hand, system operability is required for the main steam line breaks, PBOC-7 and PBOC-8, either of which could result in cumulative radiation exposures to the pump well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of P229, as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS2390A, PS2390B
TER No. N/A
Sheet 1 of 2

Preparer: NR Ein Date: 7/30/84

Independent Review: JL Rogers Date: 7/30/84

Approval: RC Denny Date: 7/30/84

These pressure switches provide a condensate storage tank level signal to HPCI valve control to open MO2301-35, 36. MO2301-35, 36 also serve as HPCI isolation valves. The HPCI isolation signal overrides the Condensate Storage Tank level signal provided by these pressure switches. Therefore, PS2390A, B are required solely to assure satisfactory HPCIS operation.

The HPCIS is relied upon to operate during and following

- Loss of Feedwater Flow,
- Total Loss of Offsite Power,
- Shutdown from Outside Control Room (Special Event),
- Pipe Break Inside Primary Containment,
- Control Rod Drop Accident, and
- Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of these pressure switches are the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS2390A, PS2390B
TER No. N/A

Preparer:	NR Ersin	Date:	7/30/84
Independent Review:	JL Rogers	Date:	7/30/84
Approval:	RA Denny	Date:	7/30/84

On the other hand, system operability is required for the main steam line breaks, PBOC-1 and PBOC-8, either of which could result in cumulative radiation exposures to the panels well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of PS2390A, B as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C2303, T2303
TER No. N/A

Sheet 1 of 2

Preparer: MR Erwin Date: 7/24/84

Independent Review: W A Clancy Date: 7/24/84

Approval: RCA Delaney Date: 7/27/84

These panels contribute to HPCIS turbine start-up and speed control and are, therefore, required solely to assure satisfactory HPCIS operation.

The HPCIS is relied upon to operate during and following

- Loss of Feedwater Flow,
- Total Loss of Offsite Power,
- Shutdown from Outside Control Room (Special Event),
- Pipe Break Inside Primary Containment,
- Control Rod Drop Accident, and
- Pipe Break Outside Primary Containment

to assure continued core cooling, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. The fourth event is addressed in the HPCIS Safety Evaluation, which states that, "The HPCIS is designed to provide adequate core cooling for small breaks...core never uncovers and is continuously cooled throughout the transient so that no core damage of any kind occurs for breaks that lie within the range of the HPCI." Thus, the size of LOCA presumed to generate postulated core damage is beyond the capacity of HPCIS to provide core cooling.

The Control Rod Drop Accident has been evaluated and no HPCIS equipment will be subjected to pressure, temperature, radiation or humidity conditions any more severe than those experienced during normal operation.

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of these panels are the PBOC-3 and the PBOC-5. Each of these events, however, incapacitates the HPCIS. System operability is, therefore, not required for either PBOC.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C2303, T2303
TER No. N/A Sheet 2 of 2

Preparer: NR Ersin Date: 7/24/84

Independent Review: WA Clancy Date: 7/24/84

Approval: [signature] Date: 7/27/84

On the other hand, system operability is required for the main steam line breaks, PBOC-7 and PBOC-8, either of which could result in cumulative radiation exposures to the panels well in excess of 10^4 rads. These values are based conservatively on the postulated core damage of NUREG 0737 and NUREG 0588. However, FSAR analysis of the PNPS design basis Main Steam Line Break Accident indicates that, with a maximum 10.5 second MSIV closure and continued core coverage (from normal or standby systems, including HPCIS), there would be no fuel damage. Without core damage, exposures will not exceed 10^4 rads.

MSIV closure time is verified once per quarter under Technical Specification surveillance requirements. The closure time must be greater than 3 seconds and less than 5 seconds for the valve to be considered operable. For valve closure times shorter than 10.5 seconds, the postulated accident is considered less severe than that analyzed.

Core cooling systems are also verified operable periodically under plant surveillance requirements. Thus, if HPCIS must be declared inoperable as a consequence of the PBOC, then ADS, LPCI and Core Spray are all assumed to be operable to assure safe shutdown of the plant. If all core cooling systems operate as designed and tested, no fuel damage should occur.

Since the assumptions of NUREG 0737 and NUREG 0588 are considered unrealistic on this basis, failure of C2303 and T2303, as a consequence of excessive radiation exposure from the main steam line break accident, is considered highly improbable and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J623, J624, J625, J626
TER No. N/A

Preparer: NR Ern... Date: 8/1/84

Independent Review: JL Rogers Date: 8/1/84

Approval: RADenny Date: 8/2/84

These junction boxes are in the electrical circuit of the outboard MSIV control modules.

The MSIV's are relied upon to function to assure reactor vessel and primary containment isolation, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines, during the following transients:

Pressure Regulator Failure,
Loss of Feedwater Flow,
Control Rod Drop Accident,
Pipe Break Inside Primary Containment, and
Pipe Break Outside Primary Containment

None of the first three events listed above is expected to result in environmental conditions any more severe than those experienced during normal operation. Also, based on FSAR analyses and event profiles, no Pipe Break Inside Primary Containment is expected to result in conditions of pressure, temperature and humidity which are any more severe in the vicinity of these outboard MSIV's than those experienced during normal operation.

Of the latter two events listed above, the PBOC with core damage generates the most severe conditions of radiation for the control modules. However, the MSIV's will receive the automatic isolation signal within 500 milliseconds of the pipe break. This is more conservative than for the PBIC. Since no electrical equipment within the valve control modules will be required to function subsequent to closure initiation, it is highly improbable that accident doses will prevent MSIV closure for required events.

Only the PBOC-7, PBOC-8 and PBOC-9 are expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the junction boxes. These conditions are not expected in the vicinity of the respective inboard MSIV control modules. These valves are tested periodically under controlled Technical Specification surveillance requirements and therefore, there is reasonable assurance that they will perform as desired. It is therefore assumed that, should the junction boxes be made inoperable, the required containment isolation would be accomplished satisfactorily by AO 203-1A/D.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J623, J624, J625, J626
TER No. N/A Sheet 2 of 2

Preparer: NR Ein Date: 8/1/84

Independent Review: JL Rogers Date: 8/1/84

Approval: RA Denny Date: 8/2/84

Based on all of the above, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. J538, J539
TER No. N/A Sheet 1 of 1

Preparer: Norm R. Erwin Date: 6/26/84

Independent Review: W S Cleary Date: 7/5/84

Approval: [signature] Date: 7/26/84

These junction boxes are in the electrical circuit for relative humidity sensors (HS-1A, 2A, 3A, 4A, 1B, 2B, 3B, 4B). The relative humidity sensors are not required for Standby Gas Treatment System (SGTS) Operation. The normal function of the sensors is to detect high humidity in the SGTS inlet and energize relays, which in turn cause the heater relays and heaters to be energized. The humidity controls have been bypassed, so that full heater operation is initiated upon operation of the SGTS exhaust fan. Therefore, these junction boxes are not required and continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A

Preparer:	JL Rogers	Date:	7/30/84
Independent Review:	MR Ev	Date:	7/30/84
Approval:	RCDenny	Date:	7/30/84

Remote shutdown panel C150 is located in the "B" RBCCW room (zone 1.22) and is exposed to a harsh superheated steam environment (peaking at 300.9°F and 15.2 psia) following a PBOC-3 (HPCI steam line break in the HPCI pump room). The panel contains series 40 Electroswitch Control Switches and GE Model ET-16 indicating lights. Failure of these components could cause loss of control to the "D" and "E" RBCCW and salt service water pumps. A reanalysis of the PBOC-3 environment for this room is presently being performed and it is expected to indicate a significantly less harsh environment than is presently assumed. In the mean time, continued operation can be justified on the following basis.

Series 40 Electroswitch

o Temperature

Temperature tests have been successfully conducted by Electroswitch on Series 24 (Report No. 2392-2) and Series 40 (Report No. 2392-14) switches. The tests were conducted at 176°F (80°C) for 120 hours. Proper operation of the switches was verified before and after the temperature exposure. For this application the maximum accident temperature is 300.9°F which exceeds the 176°F test temperature, for 45 minutes. These switches are located inside an essentially leak tight NEMA-12 enclosure (unvented) which will cause the temperature experienced by the switches to lag the accident temperature experienced by the enclosure. Tests have been conducted by Wyle Laboratories on similar sized cabinets (except with vents) which characterized the internal temperature of the cabinets as a function of time in a LOCA environment.

Results of these tests (Wyle Report No. 44439-2) show that the internal temperature of the vented cabinets lagged the external temperature by a minimum of 50°F during the first 15 minutes of exposure to a saturated steam environment. In that test the temperature and pressure were rapidly (within approximately 10 seconds) ramped to 54 psig and 280°F (minimum) respectively. Because the pressure for this application is much less than the pressure for the test (0.5 psig versus 54 psig) and in light of the essentially leak tight nature of the enclosure, the hot environment is expected to be essentially precluded from entering the panel. It is therefore our engineering judgement that in a similar test of the unvented

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A

Preparer: JL Rogers Date: 7-30-84

Independent Review: NR Ervin Date: 7/30/84

Approval: RC Cramey Date: 7/30/84

cabinets to the same maximum temperature but significantly lower pressure that the internal temperature of the cabinet would lag the external temperature by substantially greater than the 50°F experienced in the test. Further, in the tests conducted by Wyle, varied components (examples: pressure transmitter and solenoid valve) were installed in the cabinet and their mass temperature was recorded in the test. The temperature of a typical component (pressure transmitter) mounted in the vented cabinet lagged the accident temperature by approximately 80°F after the first 15 minutes of the test. It should be noted that saturated steam blanketing of high thermal inertia components such as the cabinet, as demonstrated in Limitorque Report B0027, would result in the accident environment being the equivalent of an exposure to 215°F to the exterior of the cabinet. The interior of the cabinet would be expected to lag 50 - 80°F minimum below this. In the Electroswitch test, the switches were maintained at 176°F for 120 hours. Based on the above tests and engineering rational, it is judged that the test temperature of 176°F is comparable to the temperature which the switches would experience in the accident condition. Therefore, the switches are judged suitable for use in the temperature application.

o Humidity

The compartment within which these switches are mounted experiences 100% RH for a short time period. The humidity then decays rather quickly to a long term equilibrium of approximately 60%. Due to the essentially leak tight nature of the NEMA-12 enclosures, the switches should not experience the initial humidity spike. Maximum voltage on the switches is 120 VAC. Wyle Laboratories has tested a variety of switches and terminal blocks at humidity conditions in the range of 90% to 100% including some LOCA tests. In general, no problems have been experienced for these conditions where voltage never exceeds 120 volts unless the items experienced deformation resulting from temperature. Operation of the switches at the temperature conditions is justified in the above paragraph. Also, Electroswitch has subjected the switches to 95% RH for 96 hours, unpowered. Operation of the switches was satisfactory in functional tests conducted prior to and following the humidity test. Therefore, the switches are judged suitable for use in the humidity environment.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A

Preparer: JL Rogers Date: 7-30-84

Independent Review: NR Ervin Date: 7/30/84

Approval: RC Denny Date: 7/30/84

o Pressure

The maximum pressure which the switches would be exposed to in an accident is 15.2 psia (0.5 psig). The configuration of the switches is such that they will not entrap air or otherwise cause a pressure imbalance which would result in inadvertent actuation of the switches. Therefore the switches are judged suitable for use in this pressure environment.

o Radiation

The maximum radiation which the switches will experience is approximately 1.8 x 10^3 rads. Electroswitch Test Report No. 3030-1 documents satisfactory operation of the switches following a radiation exposure of 1 x 10^7 rads. Therefore, the switches are judged suitable for use in the radiation environment.

o Aging

Conditions of aging were evaluated using the Arrhenius technique. Based on the analysis which considered all nonmetallic materials within the switch, an estimated life in excess of 40 years was established.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A Sheet 4 of 6

Preparer: Jl Rogers Date: 7-30-84

Independent Review: NR Ervin Date: 7/30/84

Approval: RCAnany Date: 7/30/84

GE Model ET-16 Lights

o Temperature

Temperature tests have been successfully conducted by Wyle on ET-16 lights. The tests were conducted at 160°F. Proper operation of the lights was verified before and after the temperature exposure. For this application the maximum accident temperature is 300.9°F which exceeds the 160°F test temperature for 75 minutes. These lights are located inside an essentially leak tight NEMA-12 enclosure (unvented) which will cause the temperature experienced by the lights to lag the accident temperature experienced by the enclosure. Tests have been conducted by Wyle Laboratories on similar sized cabinets (except with vents) which characterized the internal temperature of the cabinets as a function of time in a LOCA environment.

Results of these tests (Wyle Report No. 44439-2) show that the internal temperature of the vented cabinets lagged the external temperature by a minimum of 50°F during the first 15 minutes of exposure to a saturated steam environment. In that test the temperature and pressure were rapidly (within approximately 10 seconds) ramped to 54 psig and 280°F (minimum) respectively. Because the pressure for this application is much less than the pressure for the test (0.5 psig versus 54 psig) and in light of the essentially leak tight nature of the enclosure, the hot environment is expected to be essentially precluded from entering the panel. It is therefore our engineering judgement that in a similar test of the unvented cabinets to the same maximum temperature but significantly lower pressure that the internal temperature of the cabinet would lag the external temperature by substantially greater than the 50°F experienced in the test. Further, in the tests conducted by Wyle, varied components (examples: pressure transmitter and solenoid valve) were installed in the cabinet and their mass temperature was recorded in the test. The temperature of a typical component (pressure transmitter) mounted in the vented cabinet lagged the accident temperature by approximately 80°F after the first 15 minutes of the test. It should be noted that saturated steam blanketing of high thermal inertia components such as the cabinet, as demonstrated in Limitorque Report

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A

Preparer: JL Rogers Date: 7-30-84

Independent Review: NR Ei[illegible] Date: 7/30/84

Approval: [illegible] Date: 7/30/84

B0027, would result in the accident environment being the equivalent of a 215° saturated steam exposure to the exterior of the cabinets. The interior of the cabinet would be expected to lag 50°F to 80°F below this. In the Wyle test, the lights were maintained at 160°F. Based on the above tests and engineering rationale, it is judged that the test temperature of 160°F is comparable to the temperature which the lights would experience in the accident condition. Therefore, the lights are judged suitable for use in the temperature application.

o Humidity

The compartment within which this cabinet is mounted experiences 100% RH for a short time period. The humidity then decays rather quickly to an equilibrium of approximately 60%. Due to the leak tight nature of the NEMA-12 enclosures, the lights should not experience the initial humidity spike. Maximum voltage on the lights is 120 VAC. Wyle Laboratories has tested a variety of lights at humidity conditions in the range of 90% to 100%. In general, no problems have been experienced for these conditions where voltage never exceeds 120 volts unless the items experienced deformation resulting from temperature. Operation of the lights at the temperature conditions is justified in the above paragraph. Therefore, the lights are judged suitable for use in the humidity environment.

o Pressure

The maximum pressure which the lights would be exposed to in an accident is 15.2 psia (0.5 psig). The configuration of the lights is such that they will not entrap air or otherwise cause a pressure imbalance which would result in a functional disparity in the lights. Therefore the lights are judged suitable for use in this pressure environment.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C150
TER No. N/A Sheet 6 of 6

Preparer: [signature] Date: 7-30-84
Independent Review: NR E[signature] Date: 7/30/84
Approval: [signature] Date: 7/30/84

o Radiation

The maximum radiation which the lights will experience is approximately 1.8×10^3 rads. Proprietary Wyle Test Report No. 45625-1A documents satisfactory operation of the lights following a radiation exposure of 2.1×10^6 rads. Therefore, the lights are judged suitable for use in the radiation environment.

Based on the above information, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Ring Tongue Terminations (<4KV) Outside Containment
TER No. N/A

Preparer: J. Rogers Date: 7/31/84

Independent Review: MR Ervin Date: 8/1/84

Approval: RAAnnay Date: 8/1/84

According to Wyle Laboratories Corrective Action Report No. 47066-TER-1, the installed ring tongue terminals include both insulted and non-insulated models from a variety of manufacturers. The insulation materials used on insulated models has not been specifically identified. The commonly used insulation materials for this application are nylon, PVC, PVF, and PVDF. Justification for continued operation is required as specific qualification tests do not exist.

Uninsulated ring tongue terminals are not susceptible to degradation or environmentally induced failure at the levels of stress produced by the environments at the Pilgrim I plant. Failure of these interfaces is a function of installation configuration and terminal design.

Insulated ring tongue terminals are supplied with an insulating material covering the barrel of the terminals. This insulation is provided to prevent bare metal from protruding beyond the terminal block or connection to which it is fastened, thus reducing the hazard of shock to personnel and a possible shorting path between adjacent terminals and equipment. At the voltage levels of these terminations, the physical presence of any of the industry standard insulating materials is sufficient to perform this function.

The environments which could cause significant insulation deterioration in the Pilgrim plant are temperature and radiation. Degradation induced by these environments takes the form of material softening, material embrittlement, increased compression set, loss of elongation capability, or cracking when subjected to bending stresses or dynamic loads. None of these degradation mechanisms will impact the physical barrier insulation capability of the materials in their static termination application.

The justification discussed above has been substantiated by the application of numerous terminal lugs in nuclear equipment qualification tests. While these tests were not specifically designed to qualify the terminals and the models do not necessarily correlate with Pilgrim installed lugs, the tests demonstrate that in typical plant environments, neither insulated nor non-insulated terminal lugs constitute a significant potential failure mechanism. Samples of tests which included representative terminals as part of the test specimen or part of the test equipment are Wyle 45603-1, Wyle 45638, Franklin C5257, Wyle 43703, Wyle 44282, Wyle 44300, Franklin C5022.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Ring Tongue Terminations (<4KV) Outside Containment
TER No. N/A

Preparer: JL Rogers Date: 7/31/84

Independent Review: NR Ervin Date: 8/1/84

Approval: Date: 8/1/84

Based on the above, continued operation with existing ring tongue terminals is justified.