BOSTON EDISON COMPANY
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WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

BECo Letter No. 84-099
July 9, 1984

Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing
Office Of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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

SUBJECT: Resolution of Safety Evaluation Reports For Environmental Qualification of Safety-Related Electrical Equipment at Pilgrim Nuclear Power Station

REFERENCES:
1) BECo letter No. 83-131 dated 5/18/83, W. D. Harrington to D. B. Vassallo
2) BECo Letter No. 83-129 dated 5/18/83, W. D. Harrington to D. B. Vassallo
3) Meeting between BECo and the NRC on May 22, 1984

Dear Sir:

On May 22, 1984, Boston Edison Company met with members of your staff (Reference 3) to discuss Boston Edison Co's proposed method of resolution for each of the deficiencies contained in the Technical Evaluation Report (TER) written by Franklin Research Center under contract to the NRC. Discussions also took place at the meeting regarding 1) Boston Edison's approach in responding to 10CFR50.49 Section (b)(1), (b)(2) & (b)(3), 2) the Pilgrim Maintenance and Surveillance Program to address equipment qualification, 3) Boston Edison's position on I&E Info. Notices 82-52 & 83-72, and 4) Justification for Continued Operation.

The purpose of this letter is to provide you with 1) documentation of the discussions held at the May 22 meeting, 2) final resolution of deficiencies for all TER equipment items including the updated resolution of items which were identified as "Evaluation in Progress" at the time of May 22nd meeting, and 3) resolution of generic deficiencies listed in Section 5 of the TER. Enclosure 1 to this letter contains the summary of the proposed resolution for each of the deficiencies in the TER. For those equipment items for which the documentation for environmental qualification is not yet completed, a justification for continued operation (JCO) is provided as enclosure 2 to this letter.

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At the May 22nd, 1984 meeting, a number of specific issues related to TER resolution were discussed and their conclusions have been incorporated into the final resolution. Equipment items that are identified in enclosure 1 as "Out of Scope" to 50.49 requirements will have traceable documentation to support such a conclusion. Such documentation is not included as part of this letter. However, it is available for your audit. Other issues such as the qualification concerns with Rockbestos Cable and Terminal Blocks in instrumentation circuits in the drywell are addressed as part of the final resolution for these items in Enclosure 1.

Generic deficiencies listed in Section 5 of the TER deal with 1) "instrumentation accuracy requirements in instrument qualification evaluation" and 2) "Why Pilgrim MSLB curve ends at 2000 seconds, while the curve is continuing to rise." The instrument accuracy requirements for each instrument is addressed as part of the instrument qualification evaluation. The results of this evaluation are documented as a line item on Pilgrim equipment qualification evaluation sheets (EQES=SCEW) which are kept in our equipment qualification file. Enclosure 4 to this letter provides you with the revised Pressure - Temperature (p-t) Profiles for both inside and outside primary containment. These curves represent the most severe conditions resulting from a postulated high energy line break and are used as the basis for BECo's equipment qualification evaluation. The temperature at the end of 2000 seconds as shown on MSLB curve is controlled by procedures to stay within the drywell design temperature limit of 281°F. Hence, for equipment qualification evaluations inside drywell, the environmental conditions created as a result of LOCA and plotted in M632 SH.16 apply. THE MSLB curve (previously submitted) should be used for information only.

As agreed in the meeting items to be environmentally qualified that have been added to the "Master List of Electrical Equipment" and not factored in the TER resolution process, will be submitted with resolutions and applicable JCO's in our next submittal on August 3, 1984.

The method of identification of electrical equipment within the scope of 10CFR50.49 paragraph (b)(1), (b)(2), & (b)(3) is described in Enclosure 3 to this letter. Assessment review to verify the conclusions made under (b)(2) will be performed.

The concerns raised in IE Notices 82-52 and 83-72 and discussed at the May 22nd meeting have been evaluated and incorporated in the resolution process. Review of IE Notice 82-52 indicates that only Item 1 (E.Q. Notice 1) is applicable to PNPS. E.Q. Notice No. 1 deals with Limitorque motor operators which were tested to a much more severe environment than to which the motor operators at PNPS will ever be subjected. Enclosure 1 to this letter provides resolutions for all Limitorque motor operators at PNPS. Under Item 11, only I.E. Notice 82-03 is applicable at PNPS. This is addressed in our current evaluation. In I.E. Notice 83-72, only E.Q. Notices 21, 22 and 24 apply to PNPS. Even though equipment addressed in E.Q. Notices 21 & 22 does exist at PNPS, the failure parameters described in these notices are much too conservative for PNPS conditions. E.Q. Notice 24 is being addressed by recommended inspections and replacement of Limitorque motor operator component parts.

On Maintenance and Surveillance Practices, your staff was informed at the May 22nd meeting by BECo that the qualification of equipment will be assured from the time its qualification is established. New equipment to be added in the plant will be evaluated for E.Q. requirements prior to its procurement and hence assuring its qualification. Trending of the equipment for possible degradation of operational characteristics is currently addressed by plant Failure & Malfunction Report Process. Vendor interface is addressed by the existing BECo programs and a centralized approach througn Vendor Technical Information Program (VETIP) enhancing interface between utilities and the NSSS Vendor, and the "as needed" interface with other vendors.

As discussed at the May 22nd meeting it is requested that supplemental SER's be issued to indicate Boston Edison's Equipment Qualification Program as described in this letter meet the requirements of 10CFR50.49 and that the deficiencies noted in the SER date April 13, 1983 are considered resolved.

We would be pleased to answer any questions you may have regarding the enclosed information.

Very truly yours,

W D Harrington

W. D. Harrington

WDH/TAV/mm

Enclosure 1: TER Resolution Matrix

Enclosure 2: Justification For Continued Operation

Enclosure 3: Methodology to identify equipment within the scope of 10CFR50.59 (b)(1), (b)(2) & (b)(3)

Enclosure 4: Pressure - Temperature Profiles
M632 SH1 - 16

PILGRIM NUCLEAR POWER STATION - S.E.R. DEFICIENCY 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, 46a, 47a, 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	Solenoid Valves ASCO/HVA-90-405-2A	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
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, 239, 240, 241, 242	Cable Kerite/FR/FR, HT/FR, HT/NS	None	Qualified: Test Report 17446-2 and Analysis Report 47066-CAB-3
107, 108, 268, 269, 109	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 S1	Inadequate documentation	Test program to be initiated 9/84 with completion expected by 3/85
113, 265, 267	Terminal Block GE/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	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
136, 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.E.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.
264, 266	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 & B0119
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, 46b, 47b, 62e, 78e, 78f, 80, 83, 84, 282	Solenoid Valve ASCO/NP8320A184E	Qualified life	Out of scope of 10CFR50.49
51, 48	Solenoid Valve ASCO/HVA90405 and WP-LB-831636	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

PILGRIM NUCLAR 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, 128	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/KIY	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	Indicating Light GE/ET-16	Exempt	Out of scope of 10CFR50.49

ENCLOSURE-2

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO220-2
TER No. 1 Sheet 1 of 2

Preparer: W S Clancy Date: 20 June 84

Independent Review: Tom R E[illegible] Date: 21 June 1984

Approval: R C Deany Date: 5 July 1984

MO220-2 operates the outboard isola ion valve for the MSIV drains. The valve is located outside containment in the steam tunnel and is normally closed during plant operation except during steam line warmup or while equalizing the pressure differentials across closed MSIVs in preparation for opening. The valve is automatically closed if low-low reactor vessel level, high steam line radiation, high main steam line space temperature, high steam line flow, low steam line pressure at the turbine inlets or high reactor vessel water level is sensed. The valve could be exposed to a harsh steam and radiation environment during a PBOC-7, 8 or 9, (steam line break in steam tunnel), or to a harsh radiation environment during any other PBOC or a PBIC.

Systematic Analysis

During a PBIC or PBOC, this valve's design function is to close to provide containment isolation and prevent the release of excessive amounts of radioactive material from the drywell. In most cases, the valve would already be shut and would simply have to remain shut (i.e., not perform an "active" function). There is no credible cause for a subsequent spurious opening caused by the harsh environment since all potential sensitive control components are located in panels 903, 904 and 941 in the control and cable spreading rooms. In the rare event of a PBIC or a PBOC other than a PBOC-7, 8, or 9, during steam line warm up or while bypassing the MSIVs for opening, the valve would have sufficient time to close prior to encountering a harsh environment.

During a PBOC-7, 8, or 9, MO220-2 is required to close to provide containment isolation preventing release of excessive amounts of radioactive material from the drywell, and to terminate the transient if the break is in the drain line. In the event that the break is in an unisolated main steam line then 220-1 and 220-2 would normally be closed and would remain closed as previously discussed. If the break were in the drain line, MO220-1 would not be immediately affected by the harsh environment and would be capable of closing.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO220-2
TER No. 1

Preparer: W S Clancy Date: 20 JUNE 84

Independent Review: Norm R. E[illegible] Date: 21 June 84

Approval: R A [illegible] Date: 5 July 84

Technical Analysis

MO220-2 is equippped with a Peerless DC motor utilizing Class "B" insulation for which limited qualification documentation is available. Limitorque qualification test report B0003 documents the testing of an actuator of similar design (but with a Peerless AC motor with Class "B" insulation rather than a DC motor) in a steam and radiation environment to 250°F, 25 psig and 2×10^7 rads. The test profiles envelope the service profiles for all populated transients except for temperature during the first minute of a PBOC-6 (main steam line break in the steam tunnel). However, the thermal inertia of the operator in a super heated steam environment, as documented in Limitorque Test Report B0027, will result in temperatures within the vital poritions of the actuator and motor which are enveloped by the qualification test. The results of Limitorque Report B0003 therefore justify the capability of Class B insulation to withstand the service environment. Limitorque Qualification Test Report B0009 documents the testing of an actuator of similar design (but with a Peerless DC motor with Class "H" rather than Class "B" insulation) in a steam and radiation environment which envelopes the service environment for all postulated transients affecting MO220-2. The results of Limitorque Test Report B0009 demonstrate the capability of the commutator and brushes of a Peerless Motor to withstand the service environment. Based on these considerations, the operability of MO220-2 is adequately assured and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M04002
TER No. 2 Sheet 1 of 1

Preparer: W S Cleary Date: 7/5/84
Independent Review: Norm R Eisen Date: 7/5/84
Approval: R A Denney Date: 7/5/84

M04002 is the operator for the Class C Containment Isolation Valve in the Reactor Building Closed Cooling Water (RBCCW) Return Line from the drywell HVAC coolers. This valve, which is located in the torus compartment, is normally open and can be manually closed to prevent the release of excessive amounts of radioactive material from the drywell.

M04002 would be exposed to a harsh radiation environment during a PBIC/LOCA. However, the LOCA would have to be of sufficient magnitude and in the proper location to result in a missile or jet impingement sufficient to sever the RBCCW piping within the containment. The failure of the RBCCW piping would be almost immediately indicated in the control room by a variety of off normal alarms for the RBCCW System. The operators could be expected to diagnose this condition and remotely close M04002 from the control room in a relatively short period of time. M04002 is qualified for a radiation exposure of 2×10^7 rads as documented in Limitorque Qualification Report B0003 and would therefore remain operable for period in excess of 30 days based on projected radiation exposures. This would allow sufficient time for diagnosis and closure to occur.

M04002 would be exposed to a harsh environment during a PBOC-5 (HPCI Break in the Torus Compartment). Although not required, M04002 would remain in the open position to provide drywell cooling and would not be actively required to function. All potentially sensitive control components are located in a mild environment and would not be affected by the PBOC.

Based on this discussion, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-63
TER No. 3 Sheet 1 of 1

Preparer:	W S Cleary	Date:	20 JUNE 84
Independent Review:	JL Rogers	Date:	21 JUNE 84
Approval:	R E Grazio	Date:	6/22/84

M01001-63 is the operator for the inboard isolation valve for RHR head spray during shutdown cooling (SDC) operation. This valve can be opened during SDC to maintain saturated conditions in the reactor vessel head during reactor cooldown in order to permit a more rapid/accelerated flooding of the vessel. However, the valve is normally shut during SDC and power operations. The valve is located in containment zone 1.30 elevation 84'. The valve can be operated remotely from the control room and will automatically close in the event that low reactor vessel level, high drywell pressure or high reactor vessel pressure is sensed.

The only safety function which this valve operator performs that can be challenged by a harsh environment is that of providing containment isolation during a PBIC or a PBOC. However, the valve need not provide an "active function" since it need only remain in the normally closed position. There is no credible means for this valve to subsequently fail open as a result of the harsh environment since all potentially sensitive control circuitry is located in panels 903 and 941 in the control and cable spreading rooms.

In the rare event that a PBIC or PBOC did occur with SDC in operation, the valve would be called upon to close. However, the environment to which it was exposed would be considerably less harsh than that associated with a similar transient starting from power operation. In this event, it is believed that this valve would be able to close well before its operability would be challenged. In addition, redundancy is provided by closure of the inboard check valve (1001-64) and the outboard isolation valve (1001-60).

Since capability has been shown for the performance of the required safety function(s) and since the valve would not be required to change states at any subsequent time, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-4
TER No. 4a Sheet 1 of 2

Preparer: W S Clancy Date: 2 JULY 84

Independent Review: Non R Ervin Date: 7/3/84

Approval: RC Denny Date: 7/5/84

MO2301-4 operates the inboard isolation valve in the steam supply line to the HPCI turbine. The valve is mounted within the drywell and is actuated open in the event that reactor vessel low-low water level or high drywell pressure is sensed. The valve is over-ridden closed in the event that a HPCI steam line break is identified by high HPCI steam line space temperature or high HPCI steam line flow. The valve is normally open during operation. Potentially sensitive control circuitry for this valve is mounted in panels 903, 939 and 941 in the control room and cable spreading room and would not be subject to a harsh environment.

FSAR section 6.5.1.2.2, Safety Evaluation for the HPCIS, describes the HPCI system as one "designed to provide adequate reactor core cooling for small breaks." On this premise, a detailed analysis concluded that the "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." During such events, MO2301-4 fulfills a safety function of opening/remaining open to supply steam to the HPCI turbine and therefore facilitate HPCI operation. However, since no core damage results from those events for which HPCIS operation is essential, those components such as MO2301-4 that are considered essential for HPCIS operation will not be exposed to radiation in excess of that experienced during normal operation. In the event that the small break PBIC exposes MO2301-4 to a harsh steam environment there is a small chance that MO2301-4 could be rendered inoperable prior to opening. However, ADS/LPCI and ADS/CS would be available for redundant protection. MO2301-4 therefore need not be demonstrated to be operable for PBIC.

In the event of a PBOC in the HPCI steam lines, 2301-4 and its paired outboard isolation valve (2301-5) are required to close to prevent the excessive loss of reactor coolant and the release of radioactive material. However, there would be sufficient time delay before the PBOC caused an environment within containment sufficient to challenge the operability of MO2301-4 thus allowing automatic closure of 2301-4 to occur due to high HPCI space temperature or high HPCI steam line flow.

During any other PBOC, HPCI would be required to operate for core cooling following isolation of the leak. PNPS FSAR analyses indicates that fuel failure would not occur during any PBOC and MO2301-4 would not be exposed to a harsh environment. The use of an overconservative source term mandated by

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-4
TER No. 4a

Preparer:	W J Cleary	Date:	2 July 84
Independent Review:	Norm R. Eisen	Date:	7/3/84
Approval:	R A Denney	Date:	7/5/84

NUREG 0588/0737 would result in predicting that this valve receive a harsh radiation exposure. However the valve would remain in the desired normally open position since potentially sensitive control components would not be affected by the harsh environment.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01301-16
TER No. 4b
Sheet 1 of 1

Preparer: W S Clancy Date: 20 JUNE 84

Independent Review: Norm R Ers— Date: 6/21/84

Approval: [signature] Date: 7/5/84

M01301-16 operates the inboard isolation valve in the steam supply to the RCIC turbine. The valve is located within containment at elevation 41 and is normally open during plant operation. The valve is opened automatically if a reactor vessel low low level is sensed and will be automatically overridden closed in the event that a RCIC steam line leak is sensed by indications of either high RCIC steam space temperature or high RCIC steam flow. The valve serves a dual safety role of supplying steam to the RCIC pump turbine following a Control Rod Drop (the only accident for which RCIC operation is credited) or to provide containment isolation and terminate a PBOC-4 (RCIC Steam Line Break in the RCIC Valve Station) or a PBOC-6 (RCIC Steam Line Break in the RCIC Pump Room). The valve operator is equpped with a Reliance electric motor which was rewound with Class "H" insulation material by the GE Apparatus Service Shop in Medford, MA 8/2/80. A comparison of the GE Class "H" rewind materials with the Reliance Class "HR" OEM materials showed the rewind materials to be similar or equivilent. M01301-16 is therefore similar to the motor operator whose qualification testing was documented in Limitorque Test Report 600376A.

Following a Control Rod Drop, RCIC is utilized to provide core cooling/makeup while depressurizing the isolated reactor vessel in preparation for establishing shutdown. However, M01301-16 will not be exposed to a harsh environment since fuel failure is not predicted.

During a PBOC-4 or a PBOC-6, M01301-16 would be exposed to increased radiation as a result of fuel failure while being required to shut to provide containment isolation and terminate the transient. However, the radiation exposures experienced by M01301-16 for any PBOC are enveloped by the qualification testing documented in Limitorque Report 600376A. In addtion, redundant isolation would be provided in all cases except a PBOC-4 by the outboard isolation valve operated by M01301-17.

Based on these considerations, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO220-1
TER No. 4c
Sheet 1 of 1

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Ei Date: 7/5/84

Approval: R C Denny Date: 7/5/84

MO220-1 operates the inboard isolation valve for the MSIV drains. The valve is located within containment (zone 1.30 at elevation 18') and is normally closed during plant operation except during steam line warmup or while equalizing the pressure differentials across closed MSIVs in preparation for opening. The valve is automatically closed if low-low reactor vessel level, high steam line radiation, high main steam line space temperature, high steam line flow, low steam line pressure at the turbine inlets or high reactor vessel water level is sensed. The valve could be exposed to a harsh steam and radiation environment during a PBIC or to a harsh radiation environment following a PBOC. The design function of MO220-1 is to close to provide containment isolation and prevent the release of excessive amounts of radioactive material from the drywell. The actuator is presently equipped with a stock replacement Reliance Electric motor with Class "RH" insulation. Limitorque Qualification Test Report B0058 and Appendix B document the qualification testing of a similar actuator with a Reliance Electric motor utilizing Class RH insulation. The qualification profile envelopes the service profile for all parameters for any postulated transient affecting MO220-1. MO220-1 is therefore expected to remain operable over its 30 day mission length.

Based on these considerations, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01201-2
TER No. 5 Sheet 1 of 1

Preparer: W. J. Clancy Date: 22 JUNE 84

Independent Review: Norm R Ervin Date: 6/22/84

Approval: R.C. Denny Date: 7/5/84

M01201-2 operates the 6" inboard isolation valve in the RWCU supply line from the reactor vessel. The valve is located within containment at elevation 48' and is normally open during plant operation. The valve is automatically closed if reactor vessel low level, SLCS initiation, high temperature in the RWCU space or high RWCU flow is sensed. M01201-2 can be exposed to a harsh environment during a PBIC or a PBOC. Since all potentially sensitive control components are located in mild environments spurious actuation of M01201-2 is not deemed credible.

During a PBIC, M01201-2 is exposed to a harsh steam and radiation environment. The valve's safety function during the transient would be to close for containment isolation and prevent the release of excessive amounts of radioactive material from the drywell. M01201-2 would also be exposed to a harsh radiation environment while being required to close during a PBOC for containment isolation and in the case of a PBOC-2B/2T to also terminate a leak from the RWCU System.

Limitorque has confirmed that this valve operator was built to the same specifications as operators tested and reported in Limitorque Qualification Test reports 600198 and 600376A. However, actuator replacement is planned for documentation purposes.

The qualification testing profiles documented in Limitorque reports 600198 and 600376A envelope the service profiles over the required mission length for all postulated transients. In addition, redundant isolation can also be shown in all cases by the series outboard valve 1201-5.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-50
TER No. 5 Sheet 1 of 2

Preparer: W A Clancy Date: 22 JUNE 84
Independent Review: Norm R Eis Date: 6/22/84
Approval: R A Clancy Date: 7/5/84

M01001-50 operates the inboard isolation valve in the RHR pump shutdown cooling (SDC) suction supply line for the recirculation system. This valve is located within containment at elevation 50' (zone 1.3). The valve serves a containment isolation function during a PBIC or PBOC. The valve also serves to allow return flow from the recirculation system to the RHR pumps during SDC operation. The valve has a 30 day mission length.

Communications with the vendor have documented that the operator, motor and brake installed on 1001-50 are similar and/or equivalent to equipment tested in Limitorque Reports 600198 and 600376A. Continued operation can therefore be justified on the following basis:

- Qualification Method

This component is qualified per Limitorque Test Reports 600198 and 600376A. The qualification method used in report 600198 is in accordance with the DOR guidelines with the exception of radiation. Report 600376A, which is in accordance with the DOR guidelines, qualifies this component for radiation.

- Temperature and Pressure

Per Limitorque test report 600198 and communications from the Limitorque Corp. and Wyle Labs, this motor operator has been successfully tested to a temperature and pressure profile which envelops the service profile for all postulated transients.

- Qualification Time

Per Limitorque Test Report 600198, this component was tested for a period of 7 days, with a test profile more severe than the service profile. The service profile returns to normal conditions within approximately 6 days. However, a degradation equivalency analysis of both the motor and switch compartment components proved the 7 day test to be more severe than the 30 day accident where the accident temperature is at or below 100°F for 692 hours. Based on this analysis, adequate margin exists to ensure that this component will continue to perform its intended function for the duration of its required mission length.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO-1001-50
TER No. 5
Sheet 2 of 2

Preparer: W S Clancy Date: 22 June 84

Independent Review: Norm R. E[illegible] Date: 6/22/84

Approval: R G Deeny Date: 7/5/84

- Radiation

Per Limitorque Test Report 600376A, this type of motor operator has been successfully tested to a radiation exposure of 2 x 10^8 rads. Based on communications from Limitorque, Test Report 600376A is applicable to this operator for radiation qualification purposes. This test was performed in accordance with DOR Guidelines. The total integrated dose for this component is less than the qualified dose.

- Aging (160°F)

Component materials of the Limitorque actuators have been identified. Evaluation of these materials has been performed per DOR Guidelines and using Arrhenius Analysis Techniques. With the exception of the lubricants, the components of the actuators are considered insensitive to aging effects at a 160°F temperature. Lubricants were previously renewed by changeout.

- Drywell High Temperature (240°F)

The age sensitive components of the Limitorque actuators (the lubricants, seals, gaskets, and jumper wires) were previously inspected and replaced as necessary. The limit switches, torque switches, terminal blocks, and terminal strips were previously inspected and verified to be as tested. The Class H motors, per Limitorque requirements, was previously inspected and meggered for operation. The limit switch gear frames were previously inspected and verified to be as tested. The limit switch compartment cover was previously inspected and judged acceptable for operation by Limitorque.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO202-5A, MO202-5B
TER No. 6

Preparer: W S Cuny Date: 22 JUNE 84

Independent Review: Norm R E Date: 6/22/84

Approval: Date: 7/5/84

MO202-5A/5B are the operators of the recirculation pump discharge isolation valves. These valves are normally open during power operation but the valve in the undamaged recirculation loop is automatically signaled shut for injection loop selection during a LPCI initiation. The valve operators include a motor and magnetic brake for which complete radiation qualification data is not available. Failure of these components could result in the valve not closing or only partially closing.

- Systematic Analysis

One of these two valves is signaled closed immediately following detection of a LOCA/PBIC from the other recirculation system loop. However, closure of the valves is only required for the extremely unlikely event of a double ended rupture of the pump suction piping. The 10CFR50.46 ECCS Acceptance Criteria is satisfied providing that the recirculation pump discharge valve in the unaffected loop closes and the LPCI injection valve on the same recirculation loop opens. The pump discharge valve in the affected loop is left open to maximize reactor vessel blowdown and accelerate recirculation system depressurization to the LPCI threshold and therefore does not need to actively function. For a complete, guillotine rupture of the pump discharge piping, the two redundant low pressure core spray subsystems would provide sufficient emergency core cooling.

It is highly unlikely that these valves will fail as a result of radiation damage. The incremental increase in accumulated radiation dose from a large break LOCA should not prevent valve closure, since the valve operates within the first minute of the accident.

- Technical Analysis

Limitorque Qualification Report 600198 and Limitorque Qualification Report 600376A describe the separate testing of a similar valve operator as well as a similar motor and magnetic brake assembly. The testing involved an irradiation of 200 megarads and exposure to a harsh steam environment for thirty days at temperatures/pressures as high as 329°F/90 psig for the first hour without deleterious effects. The Dings Company, which manufactured the brakes for Reliance Electric, has verified that the brakes were constructed using Class "H" insulation. Wyle Labs has subsequently performed a material

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M0202-5A, M0202-5B
TER No. 6

Preparer: W S Clancy Date: 22 JUNE 84

Independent Review: [signature] Date: 6/22/84

Approval: [signature] Date: 7/5/84

analysis which determined that the brake materials are similar or equivilent to those used in the motor and/or brake assemblies tested as documented in 600198/600376A. The total integrated design basis PBIC 30 day estimated integrated dose (6.6 x 10^7 Rads) is significantly less than the tested dose and the test temperature and pressure profile envelop the service profile for these components. An inspection of the switch compartment was previously performed to verify the condition of components and to replace those not meeting the standards for use within containment. All potentially age sensitive components of the operators have been evaluated using Arrhenius Analysis Techniques and with the exception of lubricants are considered to be insensitive to aging effects at 160°F. Lubricants were previously renewed by changeout. Wyle Labs has performed the necessary life/aging calculations to justify continued operation to the end of cycle 7.

Based on these consideration, continuation of operation is justified until such time as qualified replacements (which have been ordered) can be installed without impacting plant availability.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO/N-109, MO/N-113
TER No. 7, 8 Sheet 1 of 1

Preparer: NR Ervin Date: 6/6/84

Independent Review: [signature] Date: 6/14/84

Approval: [signature] Date: 6/18/84

These components are the outlet dampers for SGTS filter trains and are required to open upon a Standby Gas Treatment System initiation signal. The motor operators for the dampers were deenergized by removing the fuses and the dampers are positioned such that the required airflow of 4000 scfm is maintained. Therefore, failure of this item will not affect SGTS operation and continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-60
TER No. 9

Sheet 1 of 1

Preparer: W S Clancy Date: 20 JUNE 84

Independent Review: JL Rogers Date: 21 JUNE 84

Approval: RL Grazio Date: 6/22/84

MO1001-60 operates the outboard block valve for reactor vessel head spray during shutdown cooling. This valve is normally closed but can be opened during shutdown cooling (SDC) to maintain saturated conditions in the reactor vessel head during reactor vessel cooldown and permit a more rapid/accelerated flooding of the vessel. The valve is located outside containment in the fuel pool cooling heat exchanger room (zone 1.13) and could be exposed to a harsh radiation environment during a PBIC or PBOC.

During the occurrence of a PBIC or PBOC with SDC not in service, this valve would remain in the normally closed position since potentially sensitive control components will not be affected by a harsh environment. Although the valve might not subsequently be capable of opening to accelerate vessel flooding during SDC initiation, it is not required to be open to achieve SDC.

During the occurrence of a PBIC or PBOC with SDC in service, this valve would be automatically signaled closed upon receipt of a LPCI initiation signal to isolate SDC from the reactor vessel. Based on the full power PBOC/PBIC integrated dose estimates, approximately 10 minutes would elapse prior to this valve being exposed to a harsh radiation environment thus allowing MO1001-60 more than sufficient time to close. Although the valve would be inoperable for subsequent reinitiation of SDC, it is not required as discussed above.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-23A/23B, M01001-26A/26B
TER No. 11, 20

Sheet 1 of 1

Preparer: W J Clancy Date: 21 June 84

Independent Review: J L Rogers Date: 22 JUNE 84

Approval: Fred J. Mogolesko for RE Grazio Date: 6/27/84

M01001-23A/23B and M01001-26A/26B operate the containment isolation valves for the containment spray portion of the RHR system. M01001-23A and M01001-26A are located outside containment in the RWCU heat exchanger room (zone 1.11A). M01001-23B and M01001-26B are also located outside containment at the RCIC Valve Station. These valves are all normally closed.

These valves are expected to be remotely opened by an operator in the control room during a small break steam leak within containment to prevent exceeding the drywell design temperature. Although the valves are normally closed, it is our engineering judgement that there would be sufficient time to open these valves and actuate containment drywell spray prior to these valves being exposed to a harsh radiation environment.

During a PBOC, these valves are exposed to either a harsh steam and radiation environment or to a harsh radiation environment alone. In addition, the valves could possible be exposed to a harsh radiation environment following a control rod drop. However, in all cases, these valves are required to remain in their normally closed position and are not required to actively function. Subsequent spurious actuation of the valves is not deemed credible since all potentially sensitive control components are located in mild environments.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01400-25A, M01400-25B
TER No. 12, 10b

Preparer: W J Clancy Date: 7/5/84

Independent Review: Norm R Ei... Date: 7/5/84

Approval: R A Denny Date: 7/5/84

M01400-25A/B are the operators for the downstream/isolation valves for the core spray lines. M01400-25A is located in RWCU Heat Exchanger Compartment (Zone 1.11A) and M01400-25B is located in an open area of the reactor building at elevation 51' (Zone 1.12). Both valves are normally closed but will automatically open once reactor vessel pressure has decreased to approximately 400 psig (following manual initiation or indication of low reactor vessel water level or high drywell pressure) to allow core spray to provide a core cooling safety function. The valves can be exposed to a harsh environment during a PBIC or a PBOC. The valves are equipped with a motor and electrical brake for which complete qualification data is not available.

Over the full range of analyzed PBIC break sizes, reactor vessel pressure can be shown to decrease, either due to direct blowdown (large break) or ADS (small break) without assistance from HPCI/RCIC to 400 psig or less in 5 minutes or less. A design basis PBIC manifests a hazardous radiation environment in the area where M01400-25A/B are located within approximately 7 minutes. However, since the valves are designed to operate in 10 seconds or less, completion of the open cycle prior to exposure is adequately assured. In addition, a similar motor and brake demonstrated the capability of withstanding a 200 megarad exposure (which is well in excess of the design PBIC exposure) without deleterious effect as documented in Franklin Report F-C4411. Once the valves had opened, they are expected to remain open and available for use in long term core cooling since all potentially sensitive control components are not expected to be affected.

Both M01400-25A and M01400-25B would be affected by a harsh steam and radiation environment caused by a PBOC-21 (RWCU line break in the RWCU Heat Exchanger Room). However, the A & B LPCI train would be available to fulfill the core cooling safety function.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1400-25A, MO1400-25B
TER No. 12, 10b

Preparer: W A Cleary Date: 7/5/84

Independent Review: Norm R. Ee... Date: 7/5/84

Approval: [signature] Date: 7/5/84

Both MO1400-25A and MO1400-25B would be exposed solely to harsh radiation environments during all other PBOCs. However, both valves would be capable of achieving their intended open positions prior to a harsh exposure level being reached. In addition, the capability of a similar motor/operator combination to remain operable for exposures up to 2 x 10^8 rads was documented in F-C3441 as previously discussed.

Since protection can be demonstrated in the event of all potential harsh environments challenging these valve operators, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1400-24A, MO1400-24B
TER No. 13, 10a

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R. E... Date: 7/5/84

Approval: R C Deary Date: 7/5/84

These valves are the "upstream" outboard isolation valves in the core spray (CS) supply lines. These valves are located outside the drywell in zones 1.11A (RWCU Heat Exchanger Room) and 1.12 (open area at elevation 51) respectively and could be exposed to a harsh environment during a PBOC or PBIC. The valves are normally open and are controlled by remote manual actuation from the control room or automatic open actuation in the event that low low reactor vessel level or high drywell pressure are sensed concurrent with low reactor pressure.

The core spray system provides protection (core cooling) for large or small breaks in the nuclear system when feedwater, control rod drive water, RCIS and HPCIS are unable to maintain reactor vessel water level and, in the case of small breaks, when the ADS has lowered reactor pressure below CS pump shutoff head. During such transients, the design function of these two valves is to open or remain open to permit injection of CS. However, the valves are not required to actively function (i.e., change position) during such transients, either PBIC or PBOC, since they are normally maintained in the open position. There are no credible mechanisms for inducing a spurious closure during a PBIC or PBOC since all potentially sensitive control circuitry is mounted in panels 902, 932 or 933 in the control and cable spreading rooms. In addition redundant protection is provided for large break PBIC/PBOC by LPCI and for small breaks by ADS/LPCI. Continued operation is therefore considered to be justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01201-5, M01201-80
TER No. 15, 14
Sheet 1 of 1

Preparer: W A Olary Date: 7/5/84

Independent Review: Norm R Eis Date: 7/5/84

Approval: R A Olary Date: 7/5/84

M01201-5 operates the outboard isolation valve in the RWCU suction line from the reactor vessel. M01201-80 operates the isolation valve in the RWCU return line. Both valves are located outside containment in the RWCU heat exchanger room (zone 1.11A) and are normally open during reactor operation. Both valves are automatically signaled shut to terminate a RWCU linebreak upon detection of a high flow rate to RWCU or a high temperature in the RWCU spaces, or to provide containment isolation if low reactor vessel level is detected. These valves are exposed to a harsh steam and radiation environment during a PBOC-2T (RWCU line break in the RWCU heat exchanger room) and to a harsh radiation environment during a PBIC and all other PBOCs. In all cases, these valves are required to close and remain closed to either terminate the leak and/or establish primary containment. M01201-5 is being replaced with a qualified operator under the valve betterment program. Limitorque Report B0003 documents the qualification testing of a valve operator and motor similar to M01201-80 in a harsh steam and radiation environment that envelopes the service profile for both valve operators for all postulated transients including a PBOC-2T. M01201-80 is therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal strips were used for power cable termination (required by IE Notice 83-72). Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-5
TER No. 16 Sheet 1 of 1

Preparer: W S Clomy Date: 20 JUNE 84

Independent Review: J L Rogers Date: 20 JUNE 84

Approval: R L Grazio Date: 6/23/84

MO2301-5 operates the outboard isolation valve in the steam supply line to the HPCI turbine. The valve is located outside containment in the RHR/HPCI Valve Station (zone 1.10B) and is normally open. During a transient requiring HPCI operation, the valves function is to open and remain open over a 5 hour mission time to supply steam to the HPCI pump turbine.

The FSAR Section 6.5.1.2.2 Safety Evaluation of the HPCI System, describes the system as one "designed to provide adequate reactor core cooling for small breaks." On this premise, a detailed analysis concluded that the "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 system." Based on the prevention of core damage for those small break PBIC events requiring HPCI operation, those components that are essential to HPCIS operations, such as MO2301-5, will not be exposed to radiation during such transients in excess of levels occurring during normal operation and therefore need not be qualified for such small break PBIC transients.

The only harsh environment to which MO2301-5 is exposed while being required to function is that caused by a PBOC-1 (HPCI Line Break in the HPCI Valve Station). The design function of MO2301-5 during this transient is to close to isolate the leak. However, the inboard isolation valve (MO2301-4) inside containment will be capable of closing prior to exposure to a harsh environment to provide isolation of the leak. Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-29B
TER No. 17b Sheet 1 of 1

Preparer: WS Cleary Date: 7/5/84

Independent Review: Norm R. Ervin Date: 7/5/84

Approval: RCDenny Date: 7/5/84

MO1001-29B operates the downstream LPCI injection valve for the A Recirculation Loop. The valve is located outside containment in the HPCI Valve Station (zone 1.10B) and could be exposed to a harsh environment during a PBIC or a PBOC. The valve serves to allow or prohibit LPCI or shutdown cooling (SDC) flow to the B Loop and is normally open. However, MO1001-29B will be automatically closed if a low reactor vessel level or high drywell pressure is sensed during SDC to isolate a possible leak from the RHR/SDC system. The valve can be overridden open using a pushbutton at the operator control switch at panel 903 in the control room following isolation reset. There is no credible cause for spurious operation of MO1001-29B as a result of a harsh environment since all potentially sensitive control components are mounted in panels 903, 932 and 933 in the control and cable spreading rooms.

Limitorque Test Report B0003 documents qualification testing of a similar valve operator and motor for a harsh steam and radiation exposure (250°, 25 psig and 2×10^7 rads maximum). The qualification profile envelopes the service profile for all postulated transients affecting MO1001-29B except a PBOC-1. PBOC-1 (HPCI steamline break in the HPCI valve station) exposes MO1001-29B to a harsh super-heated steam and radiation environment. The PBOC-1 service profile for temperature (309.4°F maximum) exceeds the B0003 qualification profile (250°F maximum) for approximately 2 minutes. However, the thermal inertia of the valve operator in a super-heated steam environment, as documented in Limitorque Report B0027, would cause the vital portions of the valve operator and motor to lag sufficiently to be enveloped by the qualification profile. The qualification profiles for all other parameters envelope the corresponding PBOC-1 service profiles and MO1001-29B will therefore remain operable.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-8
TER No. 18

Sheet 1 of 2

Preparer:	W S Cluny	Date:	7/5/84
Independent Review:	Nom R Esn	Date:	7/5/84
Approval:	RC Denney	Date:	7/5/84

MO2301-8 serves two functions. For events requiring isolation of HPCI, MO2301-8 (a normally shut valve) serves a containment or pressure vessel isolation function. However, redundant containment and reactor vessel isolation is provided by valve 58B (feedwater line "B" check valve).

For events requiring HPCI operation, MO2301-8 opens to admit HPCI to the "B" feedwater line. 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-8
TER No. 18

Preparer:	W A Anny	Date:	7/5/84
Independent Review:	Non R Esm	Date:	7/5/84
Approval:	R A Anny	Date:	7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of MO2301-8 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.

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 MO2301-8 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, a harsh radiation exposure will not occur.

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 MO2301-8, 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. M01301-17
TER No. 19 Sheet 1 of 1

Preparer: W A Clancy Date: 7/5/84

Independent Review: Norm R. Erin Date: 7/5/84

Approval: RC Denny Date: 7/5/84

M01301-17 operates the outboard isolation valve in the steam supply line to the RCIC pump turbine. The valve is located outside containment in the RCIC piping room (zone 1.10A) and could be exposed to a harsh operating environment during a PBIC or a PBOC. M01301-17 is automatically signaled open if a low-low reactor vessel level is sensed and is signaled closed if a RCIC pipe break is signaled based on high RCIC turbine steam flow or high temperature in the RCIC space. The valve is normally in the open position.

During a PBIC, M01001-17 would be automatically signaled open to admit steam to the RCIC turbine. However, RCIC operation is not credited in the analysis of this transient and therefore M01301-17 need not be qualified to operate during this transient. It should be noted however, that M01001-17 would be capable of opening prior to the development of a harsh radiation environment at the valve.

During a PBOC-4 (RCIC Steam Line Break in the RCIC Valve Station) or a PBOC-6 (RCIC steam line break in the RCIC Pump Compartment), M01301-17 would be exposed to a harsh steam and radiation environment. During a PBOC-4 or PBOC-6, M01301-17 is intended to automatically close based on indication of high steam flow to the RCIC turbine or high temperature in the RCIC space to terminate the accident. However, redundant protection would be provided by automatic closure of the paired inboard "in containment" valve (1301-16) in response to the same signals. Neither valve is required to provide a safety function for any other PBOC since RCIC is not credited for any PBOC.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-47
TER No. 21

Preparer: W S Clancy Date: 21 JUNE 84

Independent Review: J L Rogers Date: 22 JUNE 84

Approval: FJ Mogolesko for RE Grazio Date: 6/27/84

MO1001-47 operates the outboard isolation valve in the line running from the recirculation system to the suctions of the RHR pumps. This line is used to provide return flow from the reactor vessel during shutdown cooling (SDC) operation. This valve is therefore normally shut unless SDC is in service. The valve provides a dual safety function. During the initial stages of a PBIC or PBOC, the valve is automatically signalled closed to provide containment isolation based on an indication of low reactor vessel level or high drywell pressure. Following termination of the transient, this valve would be opened to facilitate long-term core cooling in the SDC mode of operation of the RHR system. Although the valve was assigned a 30-day mission length, the active function of opening to establish SDC is conservatively estimated to occur within 8-10 hours following the transient. There is no credible cause for spurious actuation of this valve since all potentially sensitive control components are not expected to be affected by the harsh environment. MO1001-47 is equipped with a motor and brake for which only limited qualification documentation is available.

MO1001-47 is located outside containment at the RHR valve station (zone 1.9A). This area is exposed to a harsh radiation and steam environment during a PBOC-7 (main steam line break in the condenser bay), a PBOC-8 (main steam line break in the steam tunnel) or a PBOC-9 (RWCU break at the RHR valve station). The area would also be exposed to solely a harsh radiation environment during a PBIC or any other PBOC. However, by procedure SDC would normally be secured and the valve would merely need to remain in the normally closed position. As a result, the valve would not be required to actively function during the initial most challenging stages of a PBIC or any PBOC other than a PBOC-9. In the highly unlikely event that a RWCU line break occurred with SDC in service, (PBOC-9) MO1001-47 would be actuated closed to provide containment isolation. However, the latent energy and radiation inventory present in the primary system and core when the break occurred would be significantly less than in the analyzed design PBOC-9 event due to the lower temperature/pressure and reactor non-criticality associated with SDC operation. As a result, the environment to which MO1001-47 would be exposed during its 30-second closing cycle would be significantly less harsh than in the analyzed case. Based on this, it is our engineering judgement that MO1001-47 would be capable of closing without suffering any deleterious effects. In addition, redundant containment isolation would be provided by the inboard isolation valve (1001-50) which has been demonstrated to remain functional for this event.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-47
TER No. 21 Sheet 2 of 3

Preparer:	W A Cheny	Date:	21 JUNE 84
Independent Review:	JL Rogers	Date:	22 JUNE 84
Approval:	FJ Mogolesko for RE Grazio	Date:	6/27/84

The only other occasion wherein M01001-47 could be called upon to actively function during exposure to a harsh environment would be during the establishment of SDC approximately 8-10 hours following a PBIC or PBOC. The ability of M01001-47 to remain operable for this function can be demonstrated based on the following discussion.

Limitorque Qualification Test Report #B0003 documents the qualification testing of an actuator similar to M01001-47 except that it was equipped with a Peerless AC motor with class "B" insulation rather than a DC motor. The qualification test profile envelops the service profile for all postulated transients affecting M01001-47. The results of this report can therefore be used to demonstrate the capability of class "B" insulation to withstand the service exposure estimated for M01001-47.

Limitorque Qualification Report #B0009 documents the qualification testing of an actuator essentially similar to M01001-47 except that it was equipped with a Peerless DC motor with class "H" rather than class "B" insulation. The qualification test profile envelops the service profile for all postulated transients affecting M01001-47. The results of this report can therefore be used to demonstrate the capability of the Peerless DC commutator and brushes to withstand the estimated service exposure of M01001-47.

M01001-47 is also equipped with a Sterns magnetic brake manufactured with class "A" insulation. Wyle Labs has performed a materials analysis of the brake and has determined that the brake should remain functional if operated under the conditions expected at the RHR valve station during the 8-10 hour post accident time frame wherein establishment of SDC is anticipated. This determination is based on the ambient conditions at the time of actuator operation being bounded by the design ratings of the limiting materials and the moisture resistant nature of the brake housing.

Wyle has further determined that all of the brake materials except the Phenolic case on the coil selection switch (which has a threshold of 3.4×10^5 rads) will withstand the estimated exposure of approximately 10^6 rads, 8-10 hours following a PBIC/PBOC with core damage. However, based on a 25% damage level of 10^7 rads for this material, and the design of the switch, it is our engineering judgement that this will not impair the

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-47
TER No. 21

Preparer:	WS Oleary	Date:	21 JUNE 84
Independent Review:	JL Rogers	Date:	22 JUNE 84
Approval:	FJMogolesko for REGrazio	Date:	6/27/84

operability of the brake. In the unlikely event that the brake did fail and "lock up", Limitorque has indicated that they believe the valve operator would continue to operate (but at a slower speed) since the brakes are generally designed for the normal running torque, which is approximately 20% of the stall torque of the motor.

There is a potential that M01001-47 could be temporarily submerged during a feedwater line break in the steam tunnel. However, the transient is not deemed credible to occur under conditions wherein SDC would be in operation. Therefore, M01001-47 will be in its normally closed position during the submergence and will not be called upon to actively function until 8-10 hours after the temporary submergence has been alleviated. In addition, the ability of a somewhat similar operator to actively function while submerged was inadvertently demonstrated when the test chamber accidentally flooded during qualification testing documented in Limitorque Test Report 600376A. It is therefore our engineering judgement that this temporary submergence will not impair the ability of M01001-47 to subsequently operate to facilitate establishment of shutdown cooling.

Wyle Labs has also completed two additional expected life analyses. The first analysis indicated that the most limiting brake materials have an expected life of 120 years based on conditions at the time of expected operation. The second analysis determined that the qualification testing documented in Limitorque Reports B0003/B0009 is more severe than the accident environment to which M01001-47 is exposed.

Based on these considerations, it is our engineering judgement that M01001-47 will remain operable to fulfill its required functions for all postulated transients resulting in a harsh environment. In the highly unlikely event that M01001-47 did not remain operable and prevented the establishment of SDC, the RCIC, HPCI or core spray systems could be utilized for coolant makeup while steaming to the torus through the relief valve(s) or pump turbines to stabilize plant conditions until such time as M01001-47 could be manually opened. Based on all these considerations, continued operations is justified until a qualified replacement, which has been ordered, can be installed without impacting plant availability.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-28A, M01001-28B
TER No. 22a, 17a

Preparer:	W J Cheney	Date:	7/5/84
Independent Review:	Nom R Eson	Date:	7/5/84
Approval:	R C Cheney	Date:	7/5/84

M01001-28A/28B operate the LPCI loop injection throttle globe valves. M01001-28A is located outside containment in the RHR Valve Room (zone 1.9A) and M01001-28B is located outside containment in the RHR/HPCI Piping Room (zone 1.10B). Both valves are required to be operable (to open if demanded) to pass LPCI during a PBIC/PBOC or to be open for initiation of the RHR System in the Shutdown Cooling Mode following termination of several transients. Operation of these valves could be required during exposure to a hazardous environment as a result of a PBIC or a PBOC. Limitorque report B0003 summarizes qualification testing of similar valve operators and motors to a harsh steam and radiation environment (250°F, 25 psig and 2×10^7 rad maximum).

During a PBIC, the injection throttle valve for the intact recirculation loop would be required to open and then throttle LPCI for core cooling as well as to be open for shutdown cooling for long term core cooling following termination of this transient. The harsh environment exposure would be limited to the integrated radiation exposure over the 30 day mission length which is estimated as being 4.45×10^6 rads and 3.27×10^6 rads for M01001-28A and M01001-29B respectively. However, component operation will not be affected since both operators are qualified to a 2.0×10^7 rad exposure per Limitorque Report B0003.

During a PBOC-1 (HPCI Steam Line Break in the HPCI Valve Station) M01001-28B would be exposed to a harsh super-heated steam and radiation environment. However, the service profile for temperature (309°F maximum) only exceeds the qualification profile (250°F) from B0003 for approximately 2 minutes. The thermal inertia of the operator in a superheated steam environment as documented in Limitorque Report B0027, would cause the temperature in the vital portions of the operator and motor to lag sufficiently to be enveloped by the qualification profile. In addition, both trains of core spray would be available as redundant satisfaction of the core cooling safety function during the transient. SDC could be initiated following termination of the transient using M01001-28A (which would only be subject to a radiation exposure for which it is qualified) to facilitate SDC Discharge to the A Loop.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-28A, M01001-28B
TER No. 22a, 17a

Preparer: W S Clancy Date: 5 July 84

Independent Review: Norm R E[illegible] Date: 7/5/84

Approval: [illegible] Date: 7/5/84

During a PBOC-7, 8, or 9, M01001-28A could be exposed to a harsh superheated steam and radiation environment. However, the service profiles for PBOC-7 and PBOC-9 are enveloped by the qualification test profiles in B0003 and M01001-28A is therefore qualified for these transients. During a PBOC-8 (main steamline break in the steam tunnel) the service profile for temperature (251.8°F maximum) only exceeds the qualification profile (250°F maximum) for a few seconds. The Thermal inertia of the operator in a super-heated steam environment as documented in Limitorque Report B0027, would cause the temperature of the vital portions of the valve operator and motor to lag sufficiently to be enveloped by the qualification profile. The qualification profiles for all other variables envelope the associated service profiles and M01001-28A will remain operable. In addition, LPCI and SDC could be initiated through M01001-28B in all 3 cases since its harsh exposure would be limited to a radiation environment for which it is qualified in B0003.

It should also be noted that M01001-28A might be subject to submergence following closure in response to a feedwater line break. However, it is our engineering judgment that this will not inhibit the ability of the operator to function based on the inadvertent submergence during testing of a similar operator as documented in Limitorque Qualification Testing Report 600376A.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-29A
TER No. 22b Sheet 1 of 2

Preparer:	W S Clancy	Date:	7/5/84
Independent Review:	Norm R E...	Date:	7/5/84
Approval:	[signature]	Date:	7/5/84

M01001-29A operates the downstream LPCI injection valve for the A Loop. The valve is located outside containment at the RHR valve station (zone 1.9A) and could be exposed to a harsh environment during a PBIC, PBOC or a control rod drop. The valve serves to allow or prohibit LPCI or shutdown cooling (SDC) flow to the A Loop and is normally open. However, M01001-29A will be automatically closed if a low reactor vessel level is sensed during SDC operation to isolate a possible leak from the RHR/SDC system. The valve can be overridden open using a pushbutton at the operator control switch at panel 903 in the control room. There is no credible cause for spurious operation of M01001-29A as a result of a harsh environment since all potentially sensitive control components are mounted in panels 903, 932 and 933 in the control and cable spreading rooms.

M01001-29A includes a Reliance Electric AC motor (utilizing class HR insulation) equipped with a Dings magnetic brake. The Dings Company has verified that the brake was built with insulation class "H" materials as specified by their customer, Reliance Electric. A comparison of the materials used in the brake with those used in the motor was performed by Wyle Labs. Wyle determined that the materials used in the brake are similar or equivalent to those used in the motor. It is therefore our engineering judgment that the results of qualification testing of Limitorque operators equipped with Reliance Class "HR" and Class "H" motors, as documented in Limitorque Qualification Test Reports 600198 and 600376A are applicable to M01001-29A including the motor and brake. The temperature, pressure and humidity qualification testing profiles documented in Limitorque Report 600198 envelop the service profiles for M01001-29A for all postulated transients. In addition, the seven day test profile has been shown to be more severe than the service profiles anticipated over the 30 day mission length of this component by degradation analysis. The test dose of 2.04E8 rads gamma as documented in Limitorque Test Report 600376A, more than adequately envelops the expected service exposure of 5.34E6 rads gamma for this component. The brake system which has not been irradiated is constructed of the same or equivalent materials as the motor and therefore,

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-29A
TER No. 22b

Preparer:	W A Clancy	Date:	7/5/84
Independent Review:	Norm R Ewin	Date:	7/5/84
Approval:	RCDenny	Date:	7/5/84

continued operation of the brake is justified by similarity. The brake discs, which are constructed of asbestos with a phenolic binder, have a radiation threshold of 1.8E7 rads which envelops the requirement. Beta will be reduced by the shielding effect of the equipment enclosure so that analysis concerns are only with the gamma dose.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-21, MO1001-32
TER No. 24, 23

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R. Erim Date: 7/5/84

Approval: [signature] Date: 7/5/84

MO1001-21 and MO1001-32 operate the series isolation/stop valves in the line for discharging from the RHR System to Radwaste. The valves are normally shut except while the RHR is in torus recirculation mode and draining is in progress. If the valves failed to shut during a LPCI initiation, a portion of the LPCI flow would be diverted to the Radwaste System. The valves could be exposed to a harsh environment during a PBIC or a PBOC. The valves are located outside containment in the CRD Pump Room Mezzanine (Zone 1.8).

Limitorque Qualification Test Report #B0003 documents the qualification testing of a valve operator and motor similar in design to MO1001-32. The documented test profile envelops the MO1001-32 service profile for all transients that are postulated to affect MO1001-32. MO1001-32 is therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were utilized for power cable terminations (required by IE Notice 83-72). Since isolation is the only safety function provided by MO1001-21 and MO1001-32, redundant protection for any postulated failure of MO1001-21 would be provided by MO1001-32. Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01301-25, M01301-26
TER No. 26, 25

Sheet 1 of 1

Preparer: W S Cleary Date: 7/5/84

Independent Review: Norm R Eisin Date: 7/5/84

Approval: RCCleary Date: 7/5/84

M01301-25 and M01301-26 operate the block/isolation valves in the torus suction supply line to the RCIC turbine. These valves are located in the RCIC pump room mezzanine (zone 1.5) and are normally closed. The valves are to be manually opened if low condensate storage level or high torus suppression pool level is sensed.

M01301-25 and M01301-26 can serve a containment isolation function during a PBOC or a PBIC. However, in both cases the valves are not required to actively function since they will be maintained in their normally closed position. Subsequent spurious opening of either valve is not deemed credible since all potentially sensitive control components are located in mild environments.

The only transient for which RCIC and M01301-25/26 are required to open is a Control Rod Drop. RCIC is used following a Control Rod Drop to supply core cooling while depressurizing. Sufficient reserve volume exists with condensate storage tanks for RCIC to cooldown and depressurize the plant to the shutdown cooling threshold without transferring to torus suction. The only potential harsh environment to which M01301-25 or M01301-26 could be exposed to during this time would be from radiation from fission products released from failed fuel and entrained in the steam supplied to the RCIC turbine. However, analysis has indicated that since fuel damage is not predicted to occur, this source is insufficient to expose M01301-25 or M01301-26 to a harsh environment and as such they need not be qualified. Continued operation is therefore justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR CONTINUED OPERATION

Equipment Identification No. M02301-10
TER No. 27

Sheet 1 of 2

Preparer:	W S Cherry	Date:	7/5/84
Independent Review:	Thomas R Essig	Date:	7/5/84
Approval:	B C Denny	Date:	7/5/84

This valve provides flow from the discharge of HPCIS pump P205 to the condensate storage tanks for full flow testing of the HPCIS. Because the valve is required to open for testing only, it normally remains closed during plant operation. The opening function is not safety-related. However, if the valve is opened for testing, it must close on HPCI initiation to assure adequate cooling flow to the core. Since this is its only safety-related function, operation of M02301-10 is 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-10
TER No. 27 Sheet 2 of 2

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Es[illegible] Date: 7/5/84

Approval: RA [illegible] Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of MO2301-10 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.

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 MO2301-10 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 MO2301-10, 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. MO2301-3
TER No. 28a
Sheet 1 of 1

Preparer: W S Clamp Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: R A Denny Date: 7/5/84

MO2301-3 operates the block valve in the steam supply line to the HPCI turbine. The valve is located in the HPCI pump room (zone 1.3) and is normally closed unless HPCI is in operation. During a transient requiring HPCI operation, the valves function is to open and remain open over a 5 hour mission time to supply steam to the HPCI pump turbine.

The FSAR Section 6.5.1.2.2 Safety Evaluation of the HPCI System, describes the system as one "designed to provide adequate reactor core cooling for small breaks." On this premise, a detailed analysis concluded that the "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 system." Based on the prevention of core damage for those small break PBIC events requiring HPCI operation, those components tnat are essential to HPCIS operations, such as MO2301-3, will not be exposed to radiation during such transients in excess of levels occurring during normal operation and therefore need not be qualified for such small break PBIC transients.

During a PBOC-3 (HPCI Steam Line Break in the HPCI Pump Station) MO2301-3 would be exposed to a harsh steam and radiation environment. However, HPCI operation is not required for this transient. Instead, isolation of the leak would be accomplished by automatic closure of valve 2301-4.

During any other PBOC, HPCI would be required to operate for core cooling following isolation of the leak. PNPS FSAR analyses indicates that fuel failure would not occur during any PBOC and MO2301-3 would not be exposed to a harsh environment. Although the use of an overconservative source term mandated by NUREG 0388/0737 would result in predicting that this valve receive a harsh radiation exposure, the valve would be capable of opening prior to the exposure reaching harsh levels. The valve would remain in the desired open position since potentially sensitive control components would not be affected by the harsh environment.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-9
TER No. 28b

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R E... Date: 7/5/84

Approval: R A Oney Date: 7/5/84

This valve provides the first isolation on the discharge of HPCIS pump P205. The valve is normally maintained open and closure is only accomplished through a remote manual switch in the Main Control Room (C-903). Because containment and reactor vessel isolation is provided by valves 58B (feedwater line "B") and MO2301-8, the closing function of MO2301-9 is not safety-related. However, if the valve is closed, it must open on HPCI initiation to assure adequate cooling flow to the core. Since this is its only safety- related function, operation of MO2301-9 is 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-9
TER No. 28b

Preparer: W A Clancy Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: [signature] Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of MO2301-9 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.

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 MO2301-9 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 MO2301-9, 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. MO2301-14
TER No. 29

Sheet 1 of 2

Preparer: W S Clancy Date: 5 July 84

Independent Review: Norm R Erin Date: 7/5/84

Approval: RC Denny Date: 7/5/84

On HPCIS startup, pump P205 discharge is inadequate to defeat the effect of reactor backpressure on the injection check valves. To assure safety of the pump, a flow path is provided from the discharge line to the suppression pool. This line is then automatically isolated when flow to the core is verified by an in-line sensing device. MO2301-14 provides both the initiation and isolation of minimum flow bypass. The valve must open on a HPCIS initiation coincident with a low flow signal and must close on either a turbine trip or a high flow signal. Based on the functions of this valve, operation of MO2301-14 is 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-14
TER No. 29

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Ezin Date: 7/5/84

Approval: B C McCleary Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of MO2301-14 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.

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 MO2301-14 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 MO2301-14, 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. M02301-35, M02301-36
TER No. 31, 30

Preparer: W S Cleary Date: 7/5/84

Independent Review: Norman R Eisner Date: 7/5/84

Approval: [signature] Date: 7/5/84

M02301-35 and M02301-36 operate the block/isolation valves in the line from the Suppression Pool to the HPCI Pump Suction. These valves are located outside containment in the HPCI Pump Room (zone 1.3) and are normally closed. These valves will automatically open to supply torus water to the HPCI pumps if low condenser storage tank level or high torus water level is sensed. The valves are overridden closed in the event a HPCI Steam Line Break is sensed. All potentially sensitive control components are located in mild environments.

FSAR Section 6.5.1.2.2, Safety Evaluation for the HPCI, describes the HPCI System as one "designed to provide adequate reactor core cooling for small breaks." On this premise, a detailed analysis concluded that the "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." Based on the fact that no core damage results from those events for which HPCI operation is essential, components such as M02301-35 and M02301-36, which are considered essential to HPCI operation will not be exposed to radiation in excess of the levels experienced during normal operation. As a result, capability of these components to facilitate HPCI operation while exposed to a harsh environment need not be demonstrated.

However, M02301-35 and M02301-36 provide a second safety function of closing to provide containment isolation during a PBOC-3 (HPCI Steam Line Break in the HPCI Pump Compartment) while exposed to a harsh environment as a result of blowdown from the break. If the break occurs with both valves in their normal closed position, both valves will remain closed and this design function will be accomplished.If the break occurs while both valves are open, then M02301-35 which is equipped with a rewound motor is assumed to fail as is (open). However, an operator and motor combination similar to M02301-36 was qualified to a maximum of 250°F, 25 psig and 2 x 10^7 rads as documented in Limitorque Report B0003. Although the service profile (301°F and 16.2 psia maximum) is not enveloped by the qualification profile over the first five minutes, the thermal inertia of the operator in the superheated steam environment, as documented in Limitorque Report B0027, will result in temperature in the vital portions of the actuator and motor, that would be enveloped by the qualification profile. The radiation exposure would not impact the ability of the component to operate until well after it had

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO2301-35, MO2301-36
TER No. 31, 30 Sheet 2 of 2

Preparer:	W A Olan	Date:	7/5/84
Independent Review:	Nor R Ersi	Date:	7/5/84
Approval:	RA Doney	Date:	7/5/84

closed. It can therefore be assumed that MO2301-36 would close to provide containment isolation.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO4010A/B, MO4060A/B
TER No. 33, 38 Sheet 1 of 1

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Ersin Date: 7/5/84

Approval: R A Clancy Date: 7/5/84

MO4010A/B and MO4060A/B operate block valves in parallel paths supplying RBCCW to the "B" and "A" RHR heat exchangers respectively. These valves are located outside containment in their respective RHR/Core Spray Pump Quadrants (zones 1.1 and 1.2) and are normally closed. The control room operator is expected to open at least one of the valves associated with each RHR heat exchanger approximately 10 minutes into a design basis transient. RBCCW is supplied via these valves to the RHR System in either the LPCI, torus recirculation or shutdown cooling modes and, as a result, the valves operators have a 30 day mission time. Similar valve operator and motors were qualified for extended exposure to a steam environment (250°F and 25 psig maximum) and to radiation (2×10^7 rads) and documented in Limitorque Report B0003. MO4010A/B and MO4060A/B are therefore considered to be qualified to the profiles used in the B0003 tests pending completion of an inspection to verify that appropriate terminal blocks were utilized for terminating power leaks (required by IE notice 83-72).

During a PBIC, the only potential cause for a harsh environment exposure to these valves would be increased radiation. However, analysis has shown that the valves would not be exposed to radiation in excess of the qualified level until after their 30 day mission time had elapsed and therefore, these valves would be operable when required and are considered qualified for PBIC.

During a PBOC-5 (HPCI Steam Line Break in the Torus Compartment) MO4010A/B and MO4060A/B would be exposed to a harsh steam and radiation environment. However, the qualification test profile per B0003 envelopes the service profile and the component is considered to be qualified for PBOC.

Since MO4010A/B and MO4060A/B will remain operable over their design mission length for all possible harsh environment exposures, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01400-4A, M01400-4B
TER No. 39, 36 Sheet 1 of 1

Preparer: WA Clancy Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: RA Clancy Date: 7/5/84

M01400-4A and M01400-4B operate isolation valves in the core spray test lines that run from the discharge of the core spray pumps to the torus. The valves are located outside containment within their respective RHR and core spray quadrants (zones 1.1 and 1.2). The valves are required to close when containment spray is initiated. The valves are 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 and all other PBOCs. Limitorque Report B0003 documents the qualification testing of a similar valve operator and motor in a harsh steam and radiation environment which envelopes the service environment to which these valves are exposed for all postulated transients including a PBOC-5. M01400-4A/4B are therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were used for power cable termination (required by IE Notice 83-72). Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-36A, M01001-36B, M01001-37A, M01001-37B
TER No. 40a, 32, 40j, 37f Sheet 1 of 1

Preparer: W S Cheung Date: 7/5/84

Independent Review: Norm R Ersin Date: 7/5/84

Approval: R A Denney Date: 7/5/84

M01001-36A and M01001-36B control the block valves in the RHR injection line to the suppression pool cooling spray header. M01001-37A and M01001-37B control the block valves in the RHR injection line for suppression pool cooling. All valves are located outside containment in their respective RHR train quadrants (zones 1.1 and 1.2). All four valves are normally shut but would be required to open to initiate torus spray or torus recirculation cooling, as required, during a PBOC or PBIC. The valves have a 30 day mission time. All four valves could be exposed to a harsh steam and radiation environment during a PBOC-5 (HPCI Steam Line Break in the Torus Compartment) or to a harsh radiation environment during a PBIC or all other PBOCs. Limitorque Report B0003 documents the qualification testing of a similar valve operator and motor in a harsh steam and radiation environment that envelopes the service profile for all four valves for all postulated transients including PBOC-5. M01001-36A/37B and M01001-37A/37B are therefore considered to be qualified, pending completion of an inspection to verify that appropriate terminal blocks were used for power lead termination (required by IE Notice 83-72). Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01400-3A, M01400-3B
TER No. 40b, 37g

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Erwin Date: 7/5/84

Approval: R A Denny Date: 7/5/84

M01400-3A and M01400-3B operate the isolation valves in the core spray suction lines from the suppression pool. The valves are located outside containment within their respective RHR and core spray quadrants (zones 1.1 and 1.2). The valves are required to remain functional over a 30 day mission time to facilitate core spray system operation during a PBIC or a PBOC. The valves are 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 and all other PBOCs. Limitorque Report B0003 documents the qualification testing of a similar valve operator and motor in a harsh steam and radiation environment which envelopes the service environment to which these valves are exposed for all postulated transients including a PBOC-5. M01400-3A/3B are therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were used for power cable termination (required by IE Notice 83-72). Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-7A, M01001-7B, M01001-7C, M01001-7D
TER No. 40c, 37a, 40d, 37b

Preparer: W.S. Clancy Date: 7/5/84

Independent Review: Norm R Ervin Date: 7/5/84

Approval: RC Denny Date: 7/5/84

These motor operators are installed on the RHR Pump Suction Block Valves for RHR suction from the torus. These valves are normally key-locked open except during Shutdown Cooling (SDC) Operation. The valves are located outside containment in the RHR Pump Quadrants (zones 1.1 and 1.2), and could be exposed to a harsh environment during a large break PBIC or PBOC. Spurious operation of the valve is not deemed credible since all potentially sensitive control components are not affected. These valves could be exposed to a harsh steam and radiation environment during a PBOC-5 (HPCI Steam Line Break in the Torus) or to a harsh radiation environment following a large break PBIC or any PBOC. Limitorque Report B0003 documents qualification testing of a valve operator and motor similar to M01001-7(B-D) which envelops the service exposure to these valve operators for any postulated transient. M01001-7(B-D) are 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). M01001-7A is equipped with a Reliance Electric motor that was rewound by GE at their Apparatus Service Shop in Medford MA. GE provided a Certificate of Conformance that the motor was rewound in the same manner as was found upon receipt inspection at their facility. The motor is therefore equipped with the equivalent of the Class B insulation used during original manufacture and is essentially similar to the other motors and the qualification testing documented in Limitorque Report B0003 therefore applies. Although the test profile was only for 16 days, a degradation analysis has established that the test was more severe than the 30 day mission life exposure. In addition to this technical analysis, the following systematic analysis justifies continued operation with M01001-7A as is.

During a large break PBIC from normal operating temperature and pressure, with shutdown cooling not in service, M01001-7(A-D) would be expected to remain open to supply torus suction to the RHR pumps in LPCI mode. Since the valves would already be open, no active function would be required. In addition, since there is no credible means for spurious closure, and since core spray could provide redundant protection, exposure to a harsh environment during this transient would be inconsequential. The valves would remain in the open position to facilitate long term core cooling by LPCI and drainage from the pipe break to the torus.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-7A, M01001-7B, M01001-7C, M01001-7D
TER No. 40c, 37a, 40d, 37b   Sheet 2 of 2

Preparer: W A Danny   Date: 7/5/84

Independent Review: Norm R Essin   Date: 7/5/84

Approval: RA Denny   Date: 7/5/84

During an intermediate or small break PBIC from normal operating temperature or pressure, ADS, HPCI or RCIC would actuate to depressurize the reactor vessel without core damage. M01001-7(A-D) would either remain in the open position to provide LPCI following ADS operation or to support torus recirculation cooling. However, since core damage would not occur these valves would not be exposed to a harsh environment and would remain operable.

During a PBIC of any size during SDC operation (with the lower temperatures and pressures and reactor sub-criticality necessary to support SDC operation), the environment to which M01001-7(A-D) would be exposed would be significantly less harsh and would allow sufficient time for the valves to be opened to provide LPCI. In addition, core spray would be used to provide redundant assurance of core cooling.

During a PBOC-5 (HPCI Steam Line Break in the Torus), M01001-7(A-D) could be exposed to a harsh environment. If the plant was at normal temperature and pressure, the valves would be expected to remain open to support LPCI from the torus. IF SDC was in service, the HPCI Steam Line would be isolated due to low pressure thus prohibiting the transient. In the event that M01001-7A could not be closed following termination of LPCI, long term core cooling could be provided following termination of the transient using train B of the RHR System.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-43(A-D)
TER No. 40f, 37c, 40e, 37d Sheet 1 of 1

Preparer: W A Clancy Date: 7/5/84

Independent Review: Norm R E... Date: 7/5/84

Approval: R A ... Date: 7/5/84

MO1001-43(A-D) operate the RHR Pump Shutdown Cooling (SDC) Block Valves. These valves are normally closed unless SDC is in operation. The valves are located in their respective Core Spray/RHR pump rooms (zones 1.1 and 1.2). Limitorque Report B0003 documents qualification testing of a valve operator and motor similar to MO1001-43(B-D) for a steam and radiation environment that envelops the exposure of MO1001-43(B-D) for all postulated transients. MO1001-43(B-D) are therefore considered to be qualified pending an inspection to verify that appropriate terminal blocks were used for termination of the power leads (required by IE Notice 83-72). MO1001-43A is equipped with a Reliance Electric motor that was rewound by GE at their Apparatus Service Shop in Medford, MA. GE provided a certificate of conformance that the motor was rewound in the same manner as was found upon receipt inspection at their facility. The motor is therefore equipped with the equivilent of the class "B" insulation used during original manufacture and is essentially similar to the other motors and the qualification testing documented in Limitorque Report B0003 therefore applies. Although the test profile was only for 16 days, a degradation analysis established that the test was more severe than the 30 day mission life exposure.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. MO1001-16A, MO1001-16B
TER No. 40g, 37e Sheet 1 of 1

Preparer: W.S. Chang Date: 7/5/84

Independent Review: Norm R Ericson Date: 7/5/84

Approval: [signature] Date: 7/5/84

MO1001-16A and MO1001-16B operate the RHR heat exchanger bypass valves. These valves are located in their respective RHR pump quadrants (zones 1.1 and 1.2). The valves are normally closed except while operating RHR in the shutdown cooling (SDC) mode. During SDC operation, these valves are in a throttled-open position to control reactor vessel temperature. During a LPCI initiation, both valves will be signaled open following a 60 second delay in order to maximize injection flow and control vessel cooldown. These valves are exposed to a harsh steam and radiation environment during a PBOC-5 (HPCI Steam Line Break in the Torus Compartment) or to solely a harsh radiation environment during a PBIC and all other PBOCs. The valves are required to remain functional for a 30 day mission length to facilitate LPCI flow and SDC temperature control. Limitorque Report B0003 documents the qualification of a similar operator and motor in a harsh steam and radiation environment that envelopes the service profile for both valve operators for PBIC and all PBOCs including PBOC-5. MO1001-16A and MO1001-16B are therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were used for power lead termination as required by IE Notice 83-72. Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-18A, M01001-18B
TER No. 40h, 35 Sheet 1 of 1

Preparer: W.S. Clancy Date: 7/5/84

Independent Review: Norman R. Ersin Date: 7/5/84

Approval: R.A. Devany Date: 7/5/84

M01001-18A and M01001-18B operate the block valves in the minimum flow recirculation lines from the combined RHR pump discharge to the torus. The valves are designed to open upon sensing low flow from the pumps to prevent pump overheating and to close as RHR flow approaches 20% of rated LPCI flow in either injection line to ensure adequate delivery of LPCI during a PBIC/PBOC. The valves must remain operable for at least a 30 day mission length to provide overheating protection for the RHR pumps. The valves are located in their respective RHR quadrants (zones 1.1, 1.2) and 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 solely a harsh radiation environment during a PBIC and all other PBOC's. Limitorque Report B0003 documents the qualification of a similar motor and operator in a harsh steam and radiation environment that envelopes the service profile for all postulated transients affecting either valve including PBOC-5. M01001-18A/18B are therefore considered to be qualified pending completion of an inspection to verify that appropriate terminal blocks were used for power lead termination as required by IE Notice 83-72. Continued operation is therefore justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-34A, M01001-34B
TER No. 401, 34

Preparer: W S Cleary Date: 7/5/84

Independent Review: Norm R E... Date: 7/5/84

Approval: R C Denny Date: 7/5/84

M01001-34A and 34B operate the torus cooling/torus spray line block valves. These valves are normally closed unless RHR is in operation in the torus cooling mode. The valves are located outside containment in their respective RHR and core spray pump rooms (zones 1.1 and 1.2). Limitorque Test Report B0003 documents qualification testing of a valve operator and motor similar to M01001-34A for exposure to a harsh steam and radiation environment which envelops the expected service profiles for all postulated transients affecting M01001-34A. M01001-34A is therefore considered qualified pending completion of an insrection to verify that appropriate terminal blocks were utilized for terminating the power leads (required by IE Notice 83-72). The A train of RHR could therefore provide adequate assurance of the operability of torus cooling spray regardless of the operability of M01001-34B and the B train of torus cooling spray. However, the performance of M01001-34B can be further justified using the following systematic analysis. M01001-34B is equipped with a Peerless AC motor with class B insulation for which limited qualification data is available.

During a large break PBIC or a small break followed by ADS operation, M01001-34A/34B would be initially required to close to prevent diversion of LPCI to the torus. This would normally be accomplished by the valves remaining in their normally closed position. This can be assured since all potentially sensitive control components would not be affected by a harsh environment. If the valves were in the open position at the start of the transient, they would be automatically closed in response to low reactor vessel level and high drywell pressure signals prior to a harsh radiation environment developing at their locations. The valves would then remain closed to support initiation of normal shutdown cooling (SDC) following termination of the transient or to facilitate SDC by LPCI or core spray and drainage through the break location. If torus cooling/core spray was required, M01001-34A which is qualified as documented in Limitorque Report B0003 to 2×10^7 rads, would remain operable for a period in excess of 150 days and could be used for torus recirculation/spray via the A RHR Loop.

During a small break LOCA for which HPCI or ADS is used to depressurize the reactor, M01001-34A/34B would initially be required to be closed for the LPCI mode operation of RHR and then to subsequently open for torus cooling/spray. However, such breaks do not result in core damage and as a result, M01001-34A/34B would not be exposed to a harsh environment.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01001-34A, M01001-34B
TER No. 40i, 34 Sheet 2 of 2

Preparer: W S Cherry Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: R C Deney Date: 7/5/84

During a PBOC-5 (HPCI Steam Line Break in the torus compartment) M01001-34A/34B would be exposed to a harsh environment. However, qualification testing profiles for M01001-34A as documented in B0003 envelops the service profiles for all parameters and M01001-34A is therefore qualified as discussed previously and will function as required. If M01001-34B failed in the open position, redundant isolation of the B Loop torus spray and circulation lines could be provided by M01001-36B and M01001-37B which are qualified for the PBOC-5 service profile since their qualification testing per B0003 is bounding. If M01001-34B failed closed, torus cooling/spray could be provided as required using the A RHR train.

Based on these considerations, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. M01301-60
TER No. 41 Sheet 1 of 1

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: R A Denny Date: 7/5/84

M01301-60 operates the block valve in the minimum flow bypass line from the RCIC pump to the torus. This valve is normally closed except momentarily during RCIC pump startup and during periods of RCIC pump operation at low flow rates. The valve is located in the RCIC pump mezzanine (zone 1.5) and must remain operable to ensure proper operation of the RCIC System.

The only post-accident safety function for which RCIC is credited is that of supplying reactor core cooling and makeup and depressurizing the reactor vessel following isolation due to a Control Rod Drop. However, core damage is not predicted for a control rod drop and no harsh environment occurs.

M01301-60 also serves a containment isolation function by manually closing from the control room during a PBIC or PBOC. During a PBIC, M01301-60 would be capable of closing prior to a harsh environment exposure occurring. In the event that M01301-60 was not closed prior to a harsh environment exposure during a PBIC or during a PBOC, redundant isolation would be provided by valve 1301-47.

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

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. SV2300-9
TER No. 42

Preparer: W A [signature] Date: 7/5/84

Independent Review: [signature] Date: 7/5/84

Approval: [signature] Date: 7/5/84

The HPCIS turbine is automatically shutdown by tripping the turbine stop valve closed on any of several signals. This closure is accomplished by energizing SV2300-9 and thus relieving hydraulic pressure from the stop valve actuator. Failure of the solenoid valve to operate on demand could lead to damage of the turbine or pump while inadvertent operation could threaten the ability of HPCIS to provide adequate core cooling. Based on the functions of this valve, operation of SV2300-9 is required to assure either HPCIS equipment protection or continued satisfactory system 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. SV2300-9
TER No. 42 Sheet 2 of 2

Preparer: W S Cheng Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: RADenny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of SV2300-9 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.

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 SV2300-9 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 SV2300-9, 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. CV9068A, CV9068B
TER No. 43
Sheet 1 of 2

Preparer: W S Cleary Date: 7/6/84

Independent Review: Norm R. E... Date: 7/6/84

Approval: RC Murray Date: 7/6/84

A condensate drain pot is provided on the HPCIS turbine exhaust line near where that line penetrates the Torus (X-223). Since the drain pot collects condensate from the exhaust line downstream of the containment isolation valves (on the Torus side), separate isolation valves have been provided on the line from the drain pot to the gland seal condenser. These valves (CV9068A & B) must be energized to open. This condition will exist only in the absence of a HPCIS isolation signal if either the manual control switches are positioned to "OPEN" or LS9068 senses high level in the drain pot.

These valves serve a dual safety role. During a HPCI isolation, these valves will be deenergized closed to provide containment isolation. The most likely failure mode to be induced by harsh environment exposure at this time would be solenoid deenergization with the valves subsequently failing closed. This would result in the establishment of the required containment isolation. In the unlikely event that both valves failed by sticking open, two possible scenarios could be postulated. If the valves had failed open prior to a DBA this failure would have been indicated by anomalies in the level control of the drain pot. Therefore, the operating staff would have been expected to respond by closing the two downstream manual valves to establish containment isolation and initiate a program for manual draining of the drain pots based on level alarms and/or schedule. As a result, isolation of this penetration would already be established prior to the DBA/harsh environment. If the valves failed open during a DBA requiring isolation of the torus, the liquid inventory in the torus would provide a water seal that would preclude the loss of gaseous or airborne material from the primary containment. As a result, leakage from this one inch penetration would be limited to minute amounts of water borne materials leaking past the turbine and gland seal condenser pump and blower seals. This leakage is estimated as having insignificant impact on overall containment integrity and the ability to comply with 10CFR100 limits.

The other safety related function provided by these valves is to provide for automatic intermittent draining of the HPCI turbine exhaust line drain pots. This is accomplished to prevent the accumulation of condensation that could result in a water hammer. A "failed-open" failure of these valves would have little impact with the exception of a small increase in the gland seal condenser heat loads. A "failed-closed" failure of these valves could result

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. CV9068A, CV9068B
TER No. 43 Sheet 2 of 2

Preparer: WS Clancy Date: 7/6/84

Independent Review: [signature] Date: 7/6/84

Approval: [signature] Date: 7/6/84

in excessive condensate accumulation. However, water level in the drain pot is monitored and alarmed. If the valves failed as indicated by the alarm, prior to a DBA, the operating staff would respond by providing routine manual draining of the pots. As a result, it could be reasonably expected that accumulation of sufficient condensate to inhibit subsequent HPCI initiation would be highly unlikely. In the unlikely event that HPCI operation is inhibited, redundant protection could be provided by ADS/CS, ADS/LPCI or RCIC. The valves are not required to remain operable to support HPCI operation.

Based on these considerations, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. AO 203-1A/D
TER No. 85 Sheet 1 of 2

Preparer: Thomas R Essig Date: 7/5/84

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

Approval: R A Denny Date: 7/5/84

These valve control modules provide for hydraulic actuation of the four inboard main steam isolation valves. Each module contains two pilot solenoid valves, both of which must be deenergized to initiate MSIV closure. Failure of either valve to reposition on removal of electrical power will prevent closure of the respective MSIV. The valves are normally energized to hold hydraulic air under the MSIV operating piston.

The MSIV's are relied upon to function during

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

to assure reactor vessel and primary containment isolation, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

Neither 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 Outside Primary Containment is expected to result in conditions of pressure, temperature and humidity which are any more severe in the vicinity of these inboard MSIV's than those experienced during normal operation.

Of these latter two events and the Pipe Break Inside Primary Containment, the PBOC with core damage generates the most severe conditions of radiation for the control modules. Similar controls have been tested to a level of 3×10^7 rads. During the PBOC with core damage, cumulative exposure (plus 40 year normal dose) will not exceed this level for over 2 hours. However, the MSIV's will receive the automatic isolation signal within 500 milliseconds of the pipe break. This is more conservative than either of the other two events (although closure initiates later for the PBIC, exposures will not exceed 3×10^7 rads for over 24 hours).

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. AO 203-1A/D
TER No. 85

Preparer:	Nor R Ess	Date:	7/5/84
Independent Review:	WS Clary	Date:	7/5/84
Approval:	RA Denny	Date:	7/5/84

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 PBIC is expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of AO 203-1A/D. These conditions are not expected in the vicinity of the respective outboard MSIV control modules. These valves are tested periodically under controlled Technical Specification surveillance requirements; so that there can be reasonable assurance that they will perform as desired. It is therefore assumed that, should AO 203-1A/D be made inoperable, the required containment isolation would be accomplished satisfactorily by AO 203-2A/D.

The nonmetallic component materials in the Automatic Valve Corporation C5159 solenoid operated air valve assemblies are being replaced this outage with components made of viton. Components containing viton have been previously tested and proven to have a qualified life of greater than one refueling outage. A test program, testing similar valves, is currently in progress and is expected to be completed in early 1985. Upon completion of the test program a specific qualified life will be determined.

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

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. AO 203-2A/D
TER No. 86 Sheet 1 of 2

Preparer: Norm R Ewin Date: 7/5/84

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

Approval: R A Denny Date: 7/5/84

These valve control modules provide for hydraulic actuation of the four outboard main steam isolation valves. Each module contains two pilot solenoid valves, both of which must be deenergized to initiate MSIV closure. Failure of either valve to reposition on removal of electrical power will prevent closure of the respective MSIV. The valves are normally energized to hold hydraulic air under the MSIV operating piston.

The MSIV's are relied upon to function during

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

to assure reactor vessel and primary containment isolation, and thus mitigate consequences which could result in potential offsite exposures comparable to the 10CFR100 guidelines.

Neither 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 these latter two events and the Pipe Break Outside Primary Containment, the PBOC with core damage generates the most severe conditions of radiation for the control modules. Similar controls have been tested to a level of 3×10^7 rads. During the PBOC with core damage, cumulative exposure (plus 40 year normal dose) will never exceed this level over the 30 day period evaluated. However, the MSIV's will receive the automatic isolation signal within 500 milliseconds of the pipe break. This is more conservative than either of the other two events.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. AO 203-2A/D
TER No. 86 Sheet 2 of 2

Preparer: Norman R Erwin Date: 7/5/84

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

Approval: RA Conny Date: 15/84

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 AO 203-2A/D. 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; so that there can be reasonable assurance that they will perform as desired. It is therefore assumed that, should AO 203-2A/D be made inoperable, the required containment isolation would be accomplished satisfactorily by AO 203-1A/D.

The nonmetallic component materials in the Automatic Valve Corporation C5159 solenoid operated air valve assemblies are being replaced this outage with components made of viton. Components containing viton have been previously tested and proven to have a qualified life of greater than one refueling outage. A test program, testing similar valves, is currently in progress and is expected to be completed in early 1985. Upon completion of the test program a specific qualified life will be determined.

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

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. VAC204A, VAC204B, VAC204C, VAC204D
TER No. 92

Preparer:	NR Ervin	Date:	6/6/84
Independent Review:	J Rogers	Date:	6/14/84
Approval:	D Graziano	Date:	6/18/84

Temperature

The worst case postulated PBOC has a temperature spike to 228.7°F. Within 2.5 minutes the temperature will have decreased to 180°F, and within 10 minutes the temperature will be back down to 140°F. The motors are standard AC induction motors with class B insulation having a NEMA standard maximum continuous operating rating of 130°C (226°F). Due to the short duration of the extreme peak accident temperature and rapid decay of the accident conditions to normal, the temperature due to a PBOC should have no adverse affects on the motors.

Pressure

The worst case postulated PBOC has a pressure spike of .7 psig. Within 26 seconds the pressure will have decreased to normal atmospheric pressure. The motors are dripproof, open case motors that have no pressure retaining parts. Therefore, the pressure spike will have no adverse affects on the motors.

Humidity

During the worst case postulated PBOC the humidity is assumed to approach 100% immediately after the accident and then lower back to normal. The motors are a standard AC induction motors with class B insulation. The standard type construction is of a polyester enamel coated magnet wire which is then dipped twice in a polyester varnish after winding, and therefore the motors are suitable for moderate humidity levels. Once the motors are operating, the stator temperature rise will evaporate any moisture which may collect on the windings and preclude the buildup of additional moisture. Therefore, a PBOC will have no detrimental effects on the motors.

Radiation

The worst case postulated LOCA radiation (including the 40 year dose) is 1.15×10^7 rads. The motors are AC induction motors with standard class B insulation. The radiation limiting materials are the polyester enamel and

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. VAC204A, VAC204B, VAC204C, VAC204D
TER No. 92

Preparer: NR Ev[illegible] Date: 6/6/84

Independent Review: J Rogers Date: 6/14/84

Approval: [illegible] Date: 6/18/84

polyester varnish used as the insulating materials for the windings. Class B insulating systems for various types of motors have been shown, by testing, to be capable of withstanding 2×10^8 rads when used in this application. Therefore, the radiation due to a LOCA will have no detrimental effects on the motors.

Franklin's Research Center's determination of a deficiency in the category "Documented Evidence of Qualification" is because they did not have complete information regarding these components and the qualification documents. When Boston Edison completes the qualification of these components, the applicability of the qualification documents will be conclusively proven.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HR-1A, 2A, 3A, 4A, 1B, 2B, 3B, 4B
TER No. 97 (controller) Sheet 1 of 1

Preparer: NR Ervin Date: 6/6/84

Independent Review: J. Rogers Date: 6/14/84

Approval: R Grazio Date: 6/19/84

These relative humidity controllers are not required for Standby Gas Treatment System (SGTS) operation. The normal function of the controllers are to energize resistance heaters to control the humidity of the air stream being filtered. The humidity controls have been bypassed so that full heater operation is initiated upon operation of the SGTS exhaust fan. Therefore, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Terminations-Ring Tongue (<4KV)
TER No. 100

Preparer: Nor R Ev Date: 7/5/84

Independent Review: WS Amy Date: 7/5/84

Approval: RA Amy Date: 7/5/84

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 model 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. Terminations-Ring Tongue (<4KV)
TER No. 100

Preparer: Norm R Eson Date: 7/5/84

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

Approval: [illegible] Date: 7/5/84

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

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C152, C153, C154, C155, C156, C157, C158, C159, C163
TER No. 107, 108

Preparer: Norm R Essmann Date: 7/5/84

Independent Review: W A Clancy Date: 7/5/84

Approval: RA Denny Date: 7/5/84

- 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 238.1°F which exceeds the 160°F test temperature, however, only for 15 minutes. These lights are located inside an 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 the internal cabinet temperature lagged the external temperature by a minimum of 50°F during the first 15 minutes. 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 (0.6 psig versus 54 psig) it is judged that in a similar test to the same maximum temperature 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) lagged the accident temperature by approximately 80°F after the first 15 minutes of the test. 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 envelops the temperature which the lights would experience in the accident condition. Therefore, the lights are judged suitable for use in the temperature application.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C152, C153, C154, C155, C156, C157, C158, C159, C163
TER No. 107, 108

Preparer: Norm R. E[illegible] Date: 7/5/84

Independent Review: WA Cleary Date: 7/5/84

Approval: [illegible] Date: 7/5/84

- Humidity

These lights are never exposed to more than 80% RH. 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.

- Pressure

The maximum pressure which the lights would be exposed to in an accident is 15.3 psia (0.6 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.

- Radiation

The maximum radiation which the lights will experience is less than 1×10^6 rads (2.3×10^5 rads gamma and 6.6×10^5 rads beta) based on a specific location radiation analysis. 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. C61A, C61B
Agastat Relays

TER No. 109
Sheet 1 of 1

Preparer:	Norm R Eisen	Date:	7/5/84
Independent Review:	WS Cleary	Date:	7/5/84
Approval:	[signature]	Date:	7/5/84

Review of the control circuitry and logic diagrams for the operation of the ECCS coolers show that the Agastat relays (62-1724TDE, 62-1725TDE, 62-1824TDE and 62-1825TDE) are not required to actively function for operation of the unit coolers. Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Cable-Model PE/PVC
TER No. 110, 111, 112, 118, 119, 120, 121, 122, 123, 124, 252
Sheet 1 of 1

Preparer: JL Rogers Date: 6/21/84

Independent Review: WS Clancy Date: 21 JUNE 84

Approval: [illegible] Date: 6/22/84

This equipment consists of polyethylene insulated polyvinylchloride jacketed cable 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.

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.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCI Turbine EG-R Electro Mechanical Hydraulic Actuator
TER No. 152

Preparer: W A Cleary Date: 7/5/84

Independent Review: Norm R Esi Date: 7/5/84

Approval: B A Cleary Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCI Turbine EG-R Electro Mechanical Hydraulic Actuator
TER No. 152

Preparer: W A Cloney Date: 7/5/84

Independent Review: Norm R Ess Date: 7/5/84

Approval: RA Penny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the HPCI Turbine EG-R Electro Mechanical Hydraulic Actuator 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.

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 HPCI Turbine EG-R Electro Mechanical Hydraulic Actuator 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 the HPCI Turbine EG-R Electro Mechanical Hydraulic Actuator, 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. HPCI Turbine Control Cable Assemblies
TER No. 153

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Ersin Date: 7/5/84

Approval: [signature] Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCI Turbine Control Cable Assemblies
TER No. 153

Preparer: W S Ocany Date: 7/5/84

Independent Review: Thomas R Ezsin Date: 7/5/84

Approval: RAOcany Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the HPCI Turbine Control Cable Assemblies 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.

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 HPCI Turbine Control Cable Assemblies 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 the HPCI Turbine Control Cable Assemblies, 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. HPCI Turbine Magnetic Pickup
TER No. 154

Preparer: Date: 7/5/84

Independent Review: Date: 7/5/84

Approval: Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCI Turbine Magnetic Pickup
TER No. 154

Preparer: W A Clancy Date: 7/5/84

Independent Review: Norman R Ersin Date: 7/5/84

Approval: [signature] Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the HPCI Turbine Magnetic Pickup 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.

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 HPCI Turbine Magnetic Pickup 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 the HPCI Turbine Magnetic Pickup, 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. HPCI Turbine Ramp Generator & Signal Converter Box

TER No. 155 Sheet 1 of 2

Preparer:	W S Cleary	Date:	7/5/84
Independent Review:	[illegible]	Date:	7/5/84
Approval:	[illegible]	Date:	7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCI Turbine Ramp Generator & Signal Converter Box

TER No. 155

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norman R Ervin Date: 7/5/84

Approval: R A Denny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the HPCI Turbine Ramp Generator and Signal Converter Box 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.

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 HPCI Turbine Ramp Generator and Signal Converter Box 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 the HPCI Turbine Ramp Generator and Signal Converter Box, 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. Bias Speed Potentiometer
TER No. 156 Sheet 1 of 2

Preparer: W A Cleary Date: 7/5/84

Independent Review: Norm R Ei... Date: 7/5/84

Approval: R A Denny Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Bias Speed Potentiometer
TER No. 156 Sheet 2 of 2

Preparer: W A Canny Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: RA Denny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the Bias Speed Potentiometer 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.

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 Bias Speed Potentiometer 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 the Bias Speed Potentiometer, 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. Resistor Box
TER No. 157

Preparer: W J Clancy Date: 7/5/84

Independent Review: Norm R E[illegible] Date: 7/5/84

Approval: [illegible] Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Resistor Box
TER No. 157 Sheet 2 of 2

Preparer: [signature] Date: 7/5/84

Independent Review: [signature] Date: 7/5/84

Approval: [signature] Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the Resistor Box 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.

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 Resistor Box 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 the Resistor Box, 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. EG-M Control Box
TER No. 158 Sheet 1 of 2

Preparer: W A Cleary Date: 7/5/84

Independent Review: [signature] Date: 7/5/84

Approval: [signature] Date: 7/5/84

This device contributes to HPCIS turbine speed control 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. EG-M Control Box
TER No. 158 Sheet 2 of 2

Preparer: W J Cleary Date: 7/5/84

Independent Review: Norm R Eisin Date: 7/5/84

Approval: R A Ganny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the EG-M Control Box 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.

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 EG-M Control Box 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 the EG-M Control Box, 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. DPIS-261-2A,B,C,D,E,F,G,H,J,K,L,M,N,P,R,S
TER No. 172 Sheet 1 of 1

Preparer: [signature] Date: 6 6 84

Independent Review: [signature] Date: 6-18-84

Approval: [signature] Date: 6/19/84

High steam flow in each main steam line is sensed by four indicating type differential pressure switches which sense the pressure difference across the flow restrictor in that line. High steam flow could indicate a break in a main steam line. The main steam line high differential pressure switches effect automatic isolation of all main steam lines at a setting of approximately 140% of normal main steam flow.

These switches are located in the RCIC Quad mezzanine, elev. 2'9" on Panel C-2256. These switches are required to operate in the event of PBOC-7 (Main Steam Line Break in the Condenser Bay) and PBOC-8 (Main Steam Line Break in the Steam Tunnel). In the event of PBOC-7 and PBOC-8, the isolation signal will be generated within 500 milliseconds of the break due to high differential pressure across the main steam line flow restricters. The harsh environment on the RCIC Quad Mezzanine occurs after this required safety function has been performed for both PBOC-7 and PBOC-8. This is also true for Main Steam Line Breaks Inside Containment. Once the MSIV's are signalled to close, no failure mode of the steam flow switches can prevent or reverse main steam line isolation valve closure. Deliberate operator action is necessary to reopen these valves. Closure of the switch contacts due to a short caused by the harsh environment will result in MSIV closure which is the safe position of the MSIV's.

In addition to the differential pressure switches, low pressure at the turbine inlet will initiate MSIV closure within about 200 milliseconds after the break occurs. These switches, PS-261-30A, B, C, D are located in a mild environment. These provide a backup to the differential pressure signal caused by the break.

Therefore, since completion of the safety function prior to exposure to the accident environment is accomplished and subsequent failures of the equipment does not degrade any safety function and an alternative means of accomplishing the same safety function exists, continued operation of Pilgrim Station is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. dPIS 5040A, dPIS 5040B
TER No. 173

Sheet 1 of 1

Preparer: [signature] Date: 6/?/84

Independent Review: JL Rogers Date: 6/22/84

Approval: FJ Mogolesko for REGrazio Date: 6/26/84

The primary containment is designed for an internal pressure not more than 2 psi less than the concurrent external pressure. If the suppression chamber pressure falls more than 0.5 psi below the Reactor Building pressure, dPIS 5040A&B will open contacts to deenergize SV 5040A&B respectively. These valves will, in turn, vent air from AO 5040A&B, respectively; allowing those valves to open. Consequently, air will be allowed to pass through vacuum breakers X212A&B into the Torus to repressurize containment. Failure of the differential pressure switches to deenergize SV 5040A&B when a containment vacuum is present will, therefore, threaten containment integrity.

On the other hand, AO 5040A&B also provide containment isolation. An isolation signal is provided to assure that no operator action can energize SV 5040A&B. However, this isolation signal is in series with each of the differential pressure switches; such that isolation will not prevent vacuum relief. Failure of the differential pressure switches in a position which opens AO 5040A&B despite the existence of a containment isolation signal will, therefore, threaten a breach of primary containment.

FSAR Appendix G analysis indicates that primary containment vacuum relief is required solely as an auxiliary for primary containment during the Control Rod Drop Accident and the Pipe Break Inside Primary Containment. Neither of these events will result in harsh conditions of pressure, temperature and humidity in the vicinity of the switches. Also, the greatest expected cumulative exposure (post-LOCA plus 40 year normal) is 2.84×10^6 rads, which is less than the qualified dose of 3×10^6 rads.

The harsh environment for which this equipment must be qualified results from low probability events. Events which might reasonably be anticipated during this very limited period would lead to a less severe environment and therefore, less demanding service. There was insufficient test documentation to predict a qualified life for this component, however, we are continuing our aging evaluations for equipment and as additional components requiring periodic replacement or maintenance are identified, they will be handled on a case-by-case basis.

Based on these facts, continued operation of the plant is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. DPIS1001-79B
TER No. 176

Preparer: K.A. Denny Date: 6/7/84

Independent Review: JE Rogers Date: 6/19/84

Approval: RGrazio Date: 6/20/84

Function

To protect the RHR pumps from overheating at low flow rates, a minimum flow bypass pipeline, which routes water from the pump discharge to the suppression pool, is provided for each pair of pumps. A single motor-operated valve controls the condition of each bypass pipeline. Each minimum flow bypass valve (i.e. MO1001-18A, MO1001-18B) automatically opens upon sensing low flow in both injection lines. DPIS1001-79B is used to sense flow in Loop B for this purpose. The valves automatically close when the flow approaches 20 percent of rated LPCI flow in either injection line. Continued plant operation is justified on the following bases:

Aging

Conditions of aging were evaluated using the Arrhenius technique. Based on the analysis, which considered all non-metallic materials within the switch, an estimated life in excess of 40 years was established. This calculation supports projected operability of the differential pressure switch beyond 1986.

Pressure

The service profile for the location of this device reaches a peak of 15.3 psia, whereas the test pressure reaches a maximum of 7" H_2O (14.95 psi). The service profile is above 14.95 psia for approximately 18 seconds. Based on this fact and the weathertight construction of the instrument, in our engineering judgment no functional disparities will occur.

Radiation

DPIS1001-79B is qualified to a level of 3×10^6 rads. The levels of total integrated accident dose plus 40 year normal dose for area 1.2 are 1.15×10^7 rads for LOCA and 1.08×10^7 rads for HELB with core damage. Cumulative doses over time for these events suggest a qualified mission time of either 38 hours post-LOCA or 14 hours post-HELB. Either period is considered of adequate duration to assure proper startup of RHR in the LPCI mode following the respective event. To assure proper operation subsequent to this initial startup, a fully qualified instrument provides operators, in

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. DPIS1001-79B
TER No. 176 Sheet 2 of 2

Preparer: R.A. Denny R.A. Denny Date: 6/7/84
Independent Review: JL Rogers Date: 6/19/84
Approval: [signature] Date: 6/20/84

the Main Control Room, with indication of RHR loop flow. The operators have also been provided with remote manual control of valves MO1001-18A and MO1001-18B. Should it be evident to operators that RHR loop flow is less than normal, actions can be taken sufficiently early to preclude pump damage.

Temperature

The service profile for the location of this device is less severe than the test temperature profile. Peak service temperature of 229°F is higher than the test temperature of 212°F. However, the time duration that the service temperature is above 212°F is less than one minute. The test temperature is about 40°F higher than the service profile for the remainder of the test period (6 hours). In our engineering judgment and based on preliminary calculations for similar components, the internal temperature under the service condition should not reach the test temperature of 212°F. On this basis, the temperature profile in the test report is actually more severe than the service temperature profile.

Steam Exposure

A prototype of this component was subjected to 100% humidity for 6 hours. In our engineering judgment, this test was more severe than the environment to which this component may be subjected during an accident.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. DPIS1001-79A
TER No. 180 Sheet 1 of 2

Preparer:	R A Denny R.A. Denny	Date:	6/7/84
Independent Review:	JL Rogers	Date:	6/19/84
Approval:	RE Grazio	Date:	6/20/84

Function

To protect the RHR pumps from overheating at low flow rates, a minimum flow bypass piepline, which routes water from the pump discharge to the suppression pool, is provided for each pair of pumps. A single motor-operated valve controls the condition of each bypass pipeline. Each minimum flow bypass valve (i.e. MO1001-18A, MO1001-18B) automatically opens upon sensing low flow in both injection lines. DPIS1001-79A is used to sense flow in loop A for this purpose. The valves automatically close when the flow approaches 20 percent of rated LPCI flow in either injection line. Continued plant operation is justified on the following bases:

Aging

Conditions of aging were evaluated using the Arrhenius technique. Based on the analysis, which considered all non-metallic materials within the switch, an estimated life in excess of 40 years was established. This calculation supports projected operability of the differential pressure switch beyond 1986.

Pressure

The service profile for the location of this device reaches a peak of 15.4 psia, whereas the test pressure reaches a maximum of 7" H_2O (14.95 psi). The service profile is above 14.95 psia for approximately 18 seconds. Based on this fact and the weathertight construction of the instrument, in our engineering judgment no functional disparities will occur.

Radiation

DPIS1001-73A is qualified to a level of 3×10^6 rads. The levels of total integrated accident dose plus 40 year normal dose for area 1.1 are 1.14×10^7 rads for LOCA and 1.08×10^7 rads for HELB with core damage. Cumulative doses over time for these events suggest a qualified mission time of either 28 hours post-LOCA or 14 hours post-HELB. Either period is considered of adequate duration to assure proper startup of RHR in the LPCI mode following the respective event. To assure proper operation subsequent to this initial startup, a fully qualified instrument provides operators, in

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. DPIS1001-79A
TER No. 180 Sheet 2 of 2

Preparer:	R A Denny R.A. Denny	Date:	6/7/84
Independent Review:	J Rogers	Date:	6/19/84
Approval:	R E Grazio	Date:	6/20/84

the Main Control Room, with indication of RHR loop flow. The operators have also been provided with remote manual control of valves MO1001-18A and 18B. Should it be evident to operators that RHR loop flow is less than normal, actions can be taken sufficiently early to preclude pump damage.

Temperature

The service profile for the location of this device is less severe than the test temperature profile. Peak service temperature of 225°F is higher than the test temperature of 212°F. However, the time duration that the service temperature is above 212°F is less than one minute. The test temperature is about 40°F higher than the service profile for the remainder of the test period (6 hours). In our engineering judgment and based on preliminary calculations for similar components, the internal temperature under the service condition should not reach the test temperature of 212°F. On this basis, the temperature profile in the test report is actually more severe than the service temperature profile.

Steam Exposure

A prototype of this component was subjected to 100% humidity for 6 hours. In our engineering judgment, this test was more severe than the environment to which this component may be subjected during an accident.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1451A/B, PS1464A/B
TER No. 181(51/64 A), 208(51/64 B) Sheet 1 of 1

Preparer:	[signature]	Date: 6-11-84
Independent Review:	[signature]	Date: 6-18-84
Approval:	[signature]	Date: 6/19/84

These pressure switches provide a permissive to the ADS system logic. Automatic blowdown of the reactor vessel will not occur until indication of satisfactory low pressure ECCS operation. These pressure switches provide indication of satisfactory Core Spray system operation.

Pipe Breaks Outside Containment and Pipe Breaks Inside Containment are the only design basis events which produce a harsh environment in the areas of these switches.

ADS requires low-low reactor water level, high drywell pressure, indication of Core Spray or RHR pump discharge pressure and expiration of a 2 minute time delay relay in order to automatically actuate. For PBOC's, high drywell pressure will not occur and operator action would be necessary to maintain adequate core cooling. No failure modes associated with exposure of these switches to a PBOC produced harsh environment will prevent manual actuation of ADS. Therefore, these switches do not need to be qualified for the effects of a PBOC.

These switches have been analyzed to 1×10^6 rads. For a PBIC, radiation levels of 1×10^6 rads are reached 4 hours after the pipe break. The FSAR credits operator action only when the operator can reasonably be expected to accomplish the required action under the existing conditions. In our judgement, at 4 hours into the event, operator action to initiate ADS if required, can reasonably be assumed.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS1001-93A, B, C, D; PS1001-104A, B, C, D
TER No. 182(93A/C), 182(104A/C), 209(93B/D), 209(104B/D) Sheet 1 of 1

Preparer: JL Rogers Date: 6-11-84

Independent Review: RC Denny Date: 6-15-84

Approval: R L Hragin Date: 6/19/84

These pressure switches provide a permissive to the ADS system logic. Automatic blowdown of the reactor vessel will not occur until indication of satisfactory low pressure ECCS operation. These pressure switches provide indication of satisfactory RHR system operation.

Pipe Breaks Outside Containment and Pipe Breaks Inside Containment are the only design basis events which produce a harsh environment in the areas of these switches.

ADS requires low-low reactor water level, high drywell pressure, indication of Core Spray or RHR pump discharge pressure and expiration of a 2 minute time delay relay in order to automatically actuate. For PBOC's, high drywell pressure will not occur and operator action would be necessary to maintain adequate core cooling. No failure modes associated with exposure of these switches to a PBOC produced harsh environment will prevent manual actuation of ADS. Therefore, these switches do not need to be qualified for the effects of a PBOC.

These switches have been analyzed to 1×10^6 rads. For a PBIC, radiation levels of 1×10^6 rads are reached 4 hours after the pipe break. The FSAR credits operator action only when the operator can reasonably be expected to accomplish the required action under the existing conditions. In our judgement, at 4 hours into the event, operator action to initiate ADS if required, can reasonably be assumed.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCIS Turbine Bearing Oil Pressure Switch
TER No. 185

Preparer: W A Cleary Date: 7/5/84

Independent Review: Norm R Ersin Date: 7/5/84

Approval: R A Cleary Date: 7/5/84

This switch provides a permissive to start the HPCIS Auxiliary Oil Pump on system initiation. After about 30 seconds of automatic turbine startup, the pressure supplied by the shaft driven oil pump is sufficient and this device signals the aux oil pump to stop. Failure of this switch to permit the pump start signal will result in a failure to open the two hydraulically controlled turbine steam inlet valves, thus preventing system initiation on demand. The functions of this switch, however, 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HPCIS Turbine Bearing Oil Pressure Switch
TER No. 185

Preparer: W A Cleary Date: 7/5/84

Independent Review: Norm R Ersin Date: 7/5/84

Approval: [signature] Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the HPCIS Turbine Bearing Oil Pressure Switch 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.

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 HPCIS Turbine Bearing Oil 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 the HPCIS Turbine Bearing Oil Pressure Switch, 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. PS2368A, PS2368B
TER No. 195

Preparer: W S Clancy Date: 7/5/84

Independent Review: Norm R Eisen Date: 7/5/84

Approval: [signature] Date: 7/5/84

The HPCIS turbine is automatically shutdown by tripping the turbine stop valve closed on any of several signals. One of those signals is high turbine exhaust pressure as sensed by PS2368A and PS2368B. These switches serve their safety-related function only during HPCIS operation to assure the physical integrity of the turbine exhaust pipeline.

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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. PS2368A, PS2368B
TER No. 195 Sheet 2 of 2

Preparer: W.S. Curry Date: 7/5/84

Independent Review: Norm R. Evan Date: 7/5/84

Approval: R.A. Denny Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the 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.

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 PS2368A and PS2368B 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 either PS2368A or PS2368B 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. LIS-263-72A, LIS-263-72B, LIS-263-72C, LIS-263-72D
TE8 No. 213b, 212a, 213a, 212b Sheet 1 of 2

Preparer: W S Cleary Date: 7/5/84

Independent Review: Nor R Eis Date: 7/5/84

Approval: R C Denny Date: 7/5/84

The function of these level switches is to provide automatic initiation signals to the ECCS, RCIC and Diesel Generators on reactor water level of -49" and to trip the HPCI and RCIC turbines on reactor water level of +48". These level switches are Yarway Model 4418C. These switches are believed to be qualified with the exception of the mercury switches which are installed in this model.

The only events which result in a harsh environment at the location of these level switches are PBOC's and PBIC's.

For PBOC's, only Reactor Water Cleanup System breaks result in a harsh environment at the switch locations. The service profile for these areas reaches a peak pressure of 15.3 psig at 4.9 seconds and a peak temperature of 189.6°F at 29 seconds. The pressure transient is over at 7 seconds when the pressure has dropped to essentially atmospheric pressure. In our engineering judgment, the mercury switch will undergo no functional disparities as a result of exposure to this service profile. If the feedwater system remains in service after reactor scram, then a low-low water level of -49" will not be reached. If feedwater is not available. then reactor water level will quickly drop to -49" and ECCS initiation will result. This water level will occur prior to reaching harsh radiation levels at 10 minutes. If these switches fail and cause a trip of HPCI and RCIC on a spurious high water level signal, the operator would have at least 10 minutes to utilize ADS to blowdown the reactor vessel so that core cooling can be maintained by low pressure ECCS. With the exception of the HPCI and RCIC systems, no failure mode of these switches could result in reversal of a completed safety action or prevent the accomplishment of any other safety action.

For a PBIC, radiation levels do not significantly increase above normal levels until 10 minutes after the break has occurred. For pipe breaks that are in the range of unassisted HPCI performance, no fuel damage occurs and radiation levels do not significantly increase above normal levels. For larger pipe breaks, reactor water level will drop to -49" before radiation levels significantly increase above normal levels. In addition, high drywell pressure which will result from a PBIC will provide automatic initiation of LPCI, Core Spray, HPCI, RCIC and the Diesel Generators.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LIS-263-72A, LIS-263-72B, LIS-263-72C, LIS-263-72D
TER No. 213b, 212a, 213a, 212b Sheet 2 of 2

Preparer:	W S Cheng	Date:	7/5/84
Independent Review:	Norm R. E.	Date:	7/5/84
Approval:	R C Denny	Date:	7/5/84

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LIS-57A, LIS-57B, LIS-58A, LIS-58B
TER No. 214b, 214a, 210, 211 Sheet 1 of 2

Preparer: W S Cleary Date: 7/5/84

Independent Review: [signature] Date: 7/5/84

Approval: R A Cerny Date: 7/5/84

The function of these level switches is to provide recirculation pump trip, reactor building isolation, reactor scram and isolation of various primary containment penetrations on low reactor water level (+9"). If reactor water level drops to low-low level (-49") then they effect main steam line isolation and recirculation pump trip. These level switches are Yarway Model 4418C. These switches are believed to be qualified with the exception of the mercury switches which are installed in this model.

The only events which result in a harsh environment at the location of these level switches are Pipe Breaks Outside Containment (PBOC) and Pipe Breaks Inside Containment (PBIC).

For PBOC's, only Reactor Water Cleanup System breaks result in a harsh environment at the switch locations. Calculations indicate that a reactor water level of +9" is reached at 23 seconds after this pipe break occurs. The service profile for these areas reaches a peak pressure of 15.3 psig at 4.9 seconds and a peak temperature of 189.6°F at 29 seconds. The pressure transient is over at 7 seconds when the pressure has dropped to essentially atmospheric pressure. In our engineering judgment, the mercury switch will undergo no functional disparities as a result of exposure to this service profile. If the feedwater system remains in service after reactor scram, then a low-low water level of -49" will not be reached. If feedwater is not available, then reactor water level will quickly drop to -49" and main steam line isolation will result. This water level will occur prior to reaching harsh radiation levels at 10 minutes.

In the highly unlikely event that long term exposure to the humidity inherent in PBOC causes switch failure, then spurious closure of the MSIVs could result. However, this would not occur until several hours into the transient when closure of the MSIVs following cooldown would be eminent. In addition, the operating staff would have sufficient opportunity at this point in post transient recovery, to jumper between points DD-1 to DD-2, and BB-1 to BB-2 in panel 915 in the cable spreading room and points DD-1 to DD-2 and BB-1 and BB-2 in panel 917 in the cable spreading room to eliminate these switches from these circuits.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LIS-57A, LIS-57B, LIS-58A, LIS-58B
TER No. 214b, 214a, 210, 211 Sheet 2 of 2

Preparer: W D Cheng Date: 7/5/84

Independent Review: Norm R Ervin Date: 7/5/84

Approval: RA Denny Date: 7/5/84

For a PBIC, radiation levels do not significantly increase above normal levels until 10 minutes after the break has occurred. For pipe breaks that are in the range of unassisted HPCI performance, no fuel damage occurs and radiation levels do not significantly increase above normal levels. For larger pipe breaks, reactor water level will drop to -49" before radiation levels significantly increase above normal levels. In addition, high drywell pressure will result from PBIC's and quickly effect reactor scram. As a backup to MSIV closure, if fuel damage occurs, the main steam line radiation monitors will close the MSIV's.

For both PBIC's and PBOC's, no subsequent failure modes of these switches will result in reversal of a completed safety action or prevent other safety actions from being accomplished.

Therefore, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LITS263-73A, LITS263-73B
TER No. 227 (A), 226 (B) Sheet 1 of 1

Preparer: J. Rogers Date: 6 6 84

Independent Review: B. L. Denny Date: 6-15-84

Approval: R. L. Grazio Date: 6/19/84

The function of these switches is to provide reactor water level indication in the main control room and to provide a reactor water level permissive to the containment spray subsystem of the RHR system.

The safety-related display function of these switches has been replaced by Rosemount differential pressure transmitters DPT1001-650A & B. These Rosemount transmitters Model 1153 Series B are qualified per IEEE-323-1974 and IEEE-344-1975 and the DOR guidelines to test conditions in excess of the service conditions.

The switches perform a safety-related function in a harsh environment for radiation only. The switch locations are in areas where the 40 year plus 30 day LOCA cumulative dose does not exceed 7 x 10^5 rads. The analysis which produced these radiation levels assumed that massive core damage had occurred. However, since these switches are needed only for certain small break LOCA events, it is more likely that the core will remain covered, massive core damage will not occur and radiation levels will remain mild. If these switches do fail, then the containment spray function will not be prevented. A keylocked manual override switch located in the main control room is provided to completely bypass the 2/3 core coverage permissive in the containment spray logic.

Based on these facts, continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LS2351A, LS2351B
TER No. 232 Sheet 1 of 2

Preparer: W S Cheny Date: 7/5/84

Independent Review: Norm R Ers... Date: 7/5/84

Approval: RA Denny Date: 7/5/84

These level switches provide signals to HPCIS valves MO2301-35 and MO2301-36. On high suppression pool water level, the valves are automatically opened to shift HPCIS pump suction from the condensate storage tanks to the suppression pool. Because this opening cannot occur in the presence of a system isolation signal, failure of either or both level switches will not impair the isolation function of the torus suction valves. Also, when the HPCIS is not operating, these level switches will serve no safety-related function (since suppression pool water level will not be affected by opening of the torus suction valves). These devices are therefore, required to function only during 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.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. LS2351A, LS2351B
TER No. 232

Preparer: W S Cleary   Date: 7/5/84

Independent Review: Norm R Ensin   Date: 7/5/84

Approval: [signature]   Date: 7/5/84

Those pipe breaks outside containment which could be expected to result in harsh conditions of pressure, temperature and humidity in the vicinity of the level 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.

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 LS2351A and LS2351B 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 either LS2351A or LS2351B 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. GE Cable-Model SI57275 Inside Drywell
TER No. 250 Sheet 1 of 1

Preparer: JL Rogers Date: 6/21/84

Independent Review: W S Clancy Date: 21 JUNE 84

Approval: [illegible] Date: 6/22/84

This component is GE Vulkene SIS switchboard wire which is fully qualified by test for all requirements except that the test radiation value is 4E7 rads gamma while the actual accident requirement is 6.3E7 gamma and 8.5E8 beta. Per DOR Guidelines, the minimum insulation thickness of 0.030 allows reduction of the beta dose to 8.5E7 making the total dose 14.8E7.

Franklin Institute Test report F-C2920 documents exposure of GE "Vulkene" non-jacketed single conductor cable to levels of radiation up to 5E8 gamma with subsequent LOCA testing. While not specifically referencing Model E57275, these tests were conducted prior to GE's introduction of "Vulkene Supreme" and can be considered to be generically applicable to #57275 Vulkene insulation.

This test, coupled with the actual specimen performance documented in the #57275 qualification test, is sufficient to justify continued operation.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. HS-1A, 2A, 3A, 4A, 1B, 2B, 3B, 4B
TER No. 256 Sheet 1 of 1

Preparer: TR Ervin Date: 6/5/84

Independent Review: JE Rogers Date: 6/14/84

Approval: RL Grazio Date: 6/19/84

These relative humidity sensors are not required for Standby Gas Treatment System (SGTS) Operation. The normal function of these 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, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. TSW-1A, TSW-1B
HTR T.S.

TER No. 258

Sheet 1 of 1

Preparer: Norman R. E[illegible] Date: 7/5/84

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

Approval: [illegible] Date: 7/5/84

These temperature switches provide a safety high temperature shut-off of the SGTS heaters (VGTF201A, B). They are capillary tube type of temperature switches with the following chemical compounds in the capillary tube:

1. Ortho-terphenyl 30%
2. Dipheny-ether 50%
3. Biphenyl 20%

The damage threshold of these components is at least 1 x 10^9 rads. If SGTS operated 24 hours per day post-LOCA it would take over 29 days of operation before the threshold level was reached in the SGTS charcoal beds. However, it is unlikely that SGTS will be required to operate 24 hours per day post-LOCA. Therefore, continued plant operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C68, C69
heater relay/xfmr/wire

TER No. 259, 260, 261, 262 Sheet 1 of 1

Preparer: Norm R Eisen Date: 7/5/84

Independent Review: W A Clancy Date: 7/5/84

Approval: RGDeLoney Date: 7/5/84

Transformer

The manufacturer and model listed in the Franklin TER (#260) are incorrect. The transformer was manufactured by Sola. The transformer is only required to operate post-LOCA, and is not subjected to excessive temperature and pressure. The transformer materials include kraft paper, mylar tape, cotton, and polyester; all of which have a damage threshold greater than 2×10^5 rads. The amount of radiation to which the transformer may be subjected is 1.1×10^5 rads, therefore continued plant operation is justified.

Contactor and Wire

The heater contactors (TER #261) and wire (TER #259/#262) are not required post- PBOC. They are only required to operate post-LOCA and after a fuel handling accident. A component specific calculation was performed on panels C68 and C69. The result was a worst case dose of 1.1×10^5 rads, if SGTS operated 24 hours per day post-LOCA. SGTS will probably not be required to operate continuously and therefore the actual post accident dose will be lower. Research performed by EPRI has demonstrated that with the exception of electronics, teflon, nylon fiber, and cellulose fiber, all materials reviewed had a radiation threshold level greater than the dose at the panels. There are no electronic components involved, and the nylon fiber tested was for tire cords. Cellulose fiber has a threshold of 1×10^5 rads (loss of tensile strength) but even at 4.4×10^6 rads there was only a 23% loss of tensile strength. Therefore, it would survive the postulated accident. The only remaining material that might be of concern is teflon, and it is unlikely that the material is teflon, and therefore, significant degradation of the contactor and wire is unlikely and continued operation is justified.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Electroswitch 24/40 in Alternate Shutdown Panels
TER No. 264, 266

Preparer: JL Rogers Date: 6/21/84

Independent Review: WS Cheung Date: 21 June 84

Approval: RLGrazio Date: 6/22/84

The switches are located in remote shutdown panels which provide a means of accomplishing a safe shutdown of the plant from outside the main control room. They are not required to operate in a PBOC or LOCA. However, the switches must be demonstrated to not have a failure mode during an accident which would transfer control away from the control room.

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 238.1°F which exceeds the 176°F test temperature, however, only for 15 minutes. These switches are located inside an 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 the internal cabinet temperature lagged the external temperature by a minimum of 50°F during the first 15 minutes. 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 (0.6 psig versus 54 psig) it is judged that in a similar test to the same maximum temperature 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) lagged the accident temperature by approximately 80°F after the first 15 minutes of the test. 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 envelops the temperature which the switches would experience in the accident condition. Therefore, the switches are judged suitable for use in the temperature application.

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. Electroswitch 24/40 in Alternate Shutdown Panels
TER No. 264, 266
Sheet 2 of 2

Preparer: JL Rogers Date: 6/21/84

Independent Review: W S Cleary Date: 21 June 84

Approval: R.R.Grazio Date: 6/22/84

Humidity

These switches are never exposed to more than 95% RH. Maximum voltage on the switches is 110 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 110 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.

Pressure

The maximum pressure which the switches would be exposed to in an accident is 15.3 psia (0.6 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.

Radiation

The maximum radiation which the switches will experience is less than 1×10^6 rads (2.3×10^5 rads gamma and 6.6×10^5 rads beta) based on a specific location radiation analysis. Electroswitch Test Report No. 3030-1 documents satisfactory operation of the switches following a radiation exposure of 1×10^7 rads. Therefore, the switches are judged suitable for use in the radiation environment.

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. This calculation supports projected operability of the switches well beyond 1986.

Therefore, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. C61A, C61B
Johnson Relays

TER No. 268

Sheet 1 of 1

Preparer: W J Clancy Date: 7/5/84

Independent Review: Norm R E... Date: 7/5/84

Approval: [signature] Date: 7/5/84

Review of the control circuitry and logic diagrams for the operation of the ECCS coolers show that the Johnson relays (FSE-95X, 96X, 97X, and 98X) are not required to actively function for operation of the unit coolers. Therefore, continued operation is justified.

Attachment 5 to NEDWI No. 277

BOSTON EDISON COMPANY
JUSTIFICATION FOR
CONTINUED OPERATION

Equipment Identification No. CS42-1724, CS42-1725, CS42-1824, CS42-1825
TER No. 269 Sheet 1 of 1

Preparer: NR Ev[illegible] Date: 6/6/84

Independent Review: [illegible] Date: 6/14/84

Approval: [illegible] Date: 6/20/84

The functional requirement of these switches 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 rad, 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.

ENCLOSURE 3

Compliance With 10CFR50.49

The PNPS Master Equipment List for Environmental Qualification was developed to the criteria established in 10CFR50.49 b(1), b(2), and b(3). All design basis events which could potentially result in a harsh environment were addressed in identifying safety related electrical equipment to be environmentally qualified. This assessment included all postulated events documented in Chapters 14, Appendix G, and Appendix O of the PNPS FSAR.

Section b (1) Safety-Related Equipment

Development of the Master List was performed in three phases. In the first phase, a list of systems providing a specified safety action was developed. The specified safety actions include: maintaining (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10CFR part 100 guidelines.

This phase included review of PNPS FSAR Appendices C, G and H, Safety Sequence Diagrams, and PNPS Operating Procedures. This review included all postulated design-basis accidents documented in the FSAR including a Loss of Coolant Accident (LOCA) inside containment and High Energy Line Breaks (HELB) outside containment. Flooding, pipe whip and Jet Impingement from HELB's were also analyzed.

The second Phase was to determine the specific equipment required for system operation. The documentation reviewed to determine the specific equipment required for system operation included: 1) Q-List; 2) P&ID's; 3) FSAR; 4) Technical Specifications; 5) Emergency Operating Procedures; and 6) the PNPS Cable'/Raceway Computer Program. The equipment that was excluded at this point was: 1) that which does not provide a specified safety action, 2) whose failure under postulated environmental conditions does not affect safety related equipment from performing a specified safety action, or 3) that which does not serve as post-accident monitoring equipment.

The third and final phase of the Master List development was to determine specific equipment locations and whether it was located in a harsh environment. This was determined by reviewing: 1) the EQ Project Walkdown results; 2) equipment layout drawings; 3) the PNPS Cable/Raceway Computer Program; and 4) the plant area drawings. This review was conducted so as to determine which equipment could be deleted from the Master List because that specific equipment was not located in an area of harsh environment. For equipment that was not in an area of potentially harsh environment, the cable routing was identified to assure that the cable did not pass through an area of harsh environment.

ENCLOSURE 3

Section b(2) Non-Safety Equipment Failures

Paragraph (b)(2) of 10CFR50.49 requires that licensees identify "Non-safety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions . . " Studies have been performed which address the requirements of (b)(2).

The first of these studies was in response to I&E Information Notice 79-22, dated September 14, 1979. The purpose of this study was to review non-nuclear control systems and determine if their failure due to a high energy line break could cause a safety related system to fail and thus increase the consequences of an accident. The study also evaluated whether such a failure could affect the assumptions used in the station safety analysis (FSAR Section 14).

A list of non-nuclear systems (or portions) located in an area of harsh environment, created by high energy line break was developed. A list of non-safety control systems whose failure could have an affect on a safety system or a safety analysis assumption was generated. The non-safety related equipment was considered to be of concern if its failure mode could defeat the single failure criteria or have an effect on existing safety analysis assumptions. The results of this study concluded that the reactor head vent valves could open due to a PBIC causing an increase in Peak Cladding Temperature - 10^{o}F.

The second review was performed in response to IE Bulletin 79-27 to assure that safe shutdown can be achieved in spite of single failures in safety or non-safety electric systems. In particular, the review assured that alarms or procedures exist such that failures of safety or non-safety equipment will not prevent the capability to achieve shutdown, nor will such failures lead to operator confusion in carrying out the procedures.

Third, a review of associated circuits (defined as non-safety circuits either electrically connected to safety-related circuits, located in the same raceway as safety-related circuits, or located in the same enclosure as safety related circuits) was conducted under the auspices of Appendix R. Failures and effects criteria to analyze the cables were developed. Fire-induced failures were analyzed to show that cable failure would not prevent operation or cause maloperation of systems needed for safe shutdown. Cables which could affect the safe shutdown capability of the plant will be rerouted or protected.

Boston Edison believes that a detailed review of these analyses will show that failure of non-safety related cable or non-safety related equipment will have no affect on safety related functions. An effort is currently underway at Boston Edison to complete and verify this assessment.

ENCLOSURE 3

Section b(3) "Certain Post-accident Monitoring Equipment"

The method used to identify electrical equipment within the scope of Paragraph (b)(3) of 10CFR 50.49 (i.e., "Certain post-accident monitoring equipment") involved a variable by variable comparison of the specific requirements of Regulatory Guide 1.97 to the designs of PNPS. Boston Edison projects a date of November 1984 to accomplish this effort. Any deviations found will be systematically evaluated and documented to determine if the deviation is justifiable due to plant-specific design, original design bases, or supportive operational requirements. Any deviations not found to be justifiable will be evaluated to determine what modifications, if any, are needed to conform to Regulatory Guide 1.97. Equipment that requires environmental qualification will then be identified and added on to the Master List. This equipment will be qualified in accordance with the schedule that will be established for Reg. Guide 1.97.

Attachment 1 to BECo submittal dated May 17, 1983 included certain instrumentation that is categorized as Regulatory 1.97 items. Boston Edison will endeavor to qualify this equipment according to the requirements in 10CFR 50.49. Appendix C, "Emergency Procedure Display Equipment List", included in BECo submittal dated September 11, 1981 provided the list of equipment covered under this category. However, Boston Edison did not include this equipment in attachment 1 of May 17, 1983 submittal as this list was being integrated into Regulatory Guide 1.97 effort.