CP&L

Carolina Power & Light Company

SERIAL: NLS-84-487

NOV 30 1984

Director of Nuclear Reactor Regulation
Attention: Mr. D. B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL
EQUIPMENT REVISED JUSTIFICATIONS FOR CONTINUED OPERATION

Dear Mr. Vassallo:

By letter dated November 1, 1984, Carolina Power & Light Company (the Company) was requested to review the justifications for continued operation (JCOs) for Brunswick-1 equipment qualification and confirm their acceptability on the same basis used in review of the Brunswick-2 JCOs. As a result of this review, the Brunswick-1 JCOs have been modified in order to more closely conform with review guidelines developed subsequent to their initial preparation. The revised JCOs are included as Enclosure 1.

The Company hereby confirms its belief that equipment failures which would mislead the operator or significantly degrade safety functions will not occur due to the environment resulting from a design basis event. This belief is based in part on our review of the equipment including, where appropriate, consideration of:

1. Accomplishing the safety function by some designated alternative equipment.

2. Review of partial test data.

3. Limited use of administrative controls.

4. Completion of the safety function prior to exposure of the equipment to an environment resulting from a design basis event which might degrade the equipment.

Also, the Emergency Operating Procedures provide cautions to the operator not to rely on any single indication and provides a list of instrumentation which may be used to verify the accuracy of suspect indications. The Company would like to note that, in general, JCOs have been submitted on equipment because of a lack of definitive documentation of qualification, not because the equipment is assumed to fail.

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In addition, the Company hereby confirms its belief that:

1. In performing the review of the methodology to identify equipment within the scope of 10 CFR 50.49(b)(2), the following steps were addressed:

   a. A list was generated of safety-related electric equipment as defined in paragraph (b)(1) of 10 CFR 50.49 required to remain functional during or following design-basis Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB) Accidents. The LOCA/HELB accidents are the only design-basis accidents believed to result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The list was based on reviews of the Final Safety Analysis Report (FSAR), Technical Specifications, Emergency Operating Procedures, Piping and Instrumentation Diagrams (P&IDs), and electrical distribution diagrams;

   b. The elementary diagrams of the safety-related electrical equipment identified in Step a were reviewed to identify auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment (e.g., automatic trips) whose failure due to postulated environmental conditions could prevent required operation of the safety-related equipment and;

   c. The operation of the safety-related systems and equipment were reviewed to identify directly mechanically connected auxiliary systems with electrical components which are necessary for the required operation of the safety-related equipment (e.g., cooling water or lubricating systems). This involved the review of P&IDs, component technical manuals, and/or systems descriptions in the FSAR.

   d. Nonsafety-related electrical circuits indirectly associated with the electrical equipment identified in Step a by common power supply or physical proximity were considered by a review of the electrical design including the use of applicable industry standards (e.g., IEEE, NEMA, ANSI, UL, and NEC) and the use of properly coordinated protective relays, circuit breakers, and fuses for electrical fault protection.

2. Design basis events which could potentially result in a harsh environment, including flooding outside containment, were addressed in identifying safety-related electrical equipment within the scope of 10 CFR 50.49(b)(1).

3. Electrical equipment within the scope of 10 CFR 50.49(b)(3) is R.G. 1.97 Category 1 and 2 equipment or that justification has been provided for any such equipment not included in the environmental qualification program.

Should you have any questions regarding this issue, please contact Mr. Sherwood Zimmerman (919) 836-6242.

Yours very truly,

A. B. Cutter - Vice President
Nuclear Engineering & Licensing

MAT/pgp (882MAT)
Enclosure

cc: Mr. D. O. Myers (NRC-BNP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. M. Grotenhuis (NRC)

A. B. Cutter, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Lisa M. Randall
Notary (Seal)

My commission expires: 5/18/88

LISA RANDALL
NOTARY
***
PUBLIC
WAKE COUNTY, N.C.

Enclosure 1 to NLS-84-487

Justifications for Continued Operation

JUSTIFICATION FOR CONTINUED OPERATION

EQUIPMENT LIST

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
2	17	E51-F019	Limitorque Motorized Valve Operator	(2)
3	20	B21-F016	Limitorque Motorized Valve Operator	(2)
		E11-F022		
		E51-F007		
		G31-F001		
5	24 thru 30	B32-F020-S	ASCO Solenoid	(2)
	35 thru 39	CAC-PV-1260-S		
	41,42,44	CAC-PV-1261-S		
	45,47,48	CAC-PV-1262-S		
	49,50,52	CAC-PV-3439-IS		
	53	CAC-PV-3439-S		
		CAC-PV-3440-IS		
		CAC-PV-3440-S		
		CAC-SV-4222		
		CAC-SV-4223		
		CAC-V4-S		
		CAC-V5-IS		
		CAC-V5-S		
		CAC-V6-IS		
		CAC-V6-S		
		CAC-V7-S		
		CAC-V8-S		
		CAC-V9-S		
		CAC-V10-S		
		CAC-V15-S		
		CAC-V47-IS		
		CAC-V47-S		
		CAC-V48-IS		
		CAC-V48-S		
		CAC-V49-S		
		CAC-V50-S		
		CAC-V55-S		
		CAC-V56-S		
		C11-F110A		
		C11-F110B		
		E11-F053A		
		E11-F053B		
		G16-F003-S		
		G16-F004-S		
		G16-F019-S		
		G16-F020-S		

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
5 (Cont'd)		RIP-SV-1200A1-1233B2 (150 VALVES) 1A-BFIV-RB-S 1B-BFIV-RB-S 1C-BF8V-RB-S 1D-BFIV-RB-S		
6	34,113,114, 123	VA-TS-936A thru F VA-ZS-936A,B VA-SV-936A,B	Johnson Services Temperature Switch	(1),(5)
9	62	E41-PS-N010 E51-PSL-N006	Static-O-Ring Pressure Switch	(2)
10	67	CAC-PT-1257-2	Bailey Transmitters	(1)
11	68	C32-PT-N005A,B	General Electric Pressure Transmitter	(2)
12	69	E11-PDT-N002A,B	General Electric Pressure Transmitter	(1),(5)
13	71 thru 74, 76 thru 81, 99	E11-PS-N016A thru D E11-PS-N020 A thru D RIP-PSL-1200 RIP-PSL-1201 RIP-PSL-1203 RIP-PSL-1205 RIP-PSL-1206 RIP-PS2-1208 thru 12 RIP-PSL-1215 RIP-PSL-1217, thru 23 RIP-PSL-1225 thru 29 RIP-PSL-1231 thru 32 E41-PSH-N012A thru D E41-PSH-N017A,B E51-PS-N020 E51-PSH-N009A,B E51-PSH-N012A thru D B32-PS-N018A,B B32-PS-N018A-1 SW-TSH-1109 thru 1112	Barksdale Switches	(2)

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
13 (Cont'd)		IA-PSL-3594,3595 E41-PSH-N027 SW-PS-1175,1176		
15	82	E41-LSH-N015A,B	Robertshaw Level Switches	(2)
17	91	B21-FS-F015 A thru H B21-FS-F015 J thru N B21-FS-F015P B21-FS-F015R,S B21-FS-F043A,B B21-FS-F045A,B B21-FS-F047A,B B21-FS-F049A,B B21-FS-F051A,B B21-FS-F057 B32-FS-F009C B32-FS-F040B,C B32-FS-F055E,F B32-FS-F057A,B E41-FS-F024A thru D E51-FS-F044A thru D	Magnetrol Flow Switches	(2)
18	93	VA-FT-2577	Bailey Transmitters	(2)
19	94,122	B21-F014A thru H B21-F014J thru N B21-F014P B21-F014R B21-F014S B21-F042A,B B21-F044A,B B21-F046A,B B21-F048A,B B21-F050A,B B21-F056 B32-F039B,C B32-F041C B32-F056E,F B32-F058A,B CAC-PV-1218C CAC-PV-1219B,C CAC-PV-1225C E11-F037A thru D E41-PV-1218D	Cherry Switches	(2)

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
19 (Cont'd)		E41-PV-1219D E41-PV-1220D E41-PV-1221D E41-F023A thru D E51-F043A thru D IA-PV-1201A CAC-PV-1209D E11-F043A thru D		
20	95	E41-FT-N008	General Electric Flow Transmitter	(2)
21	96,97,98	E11-PDIS-N021A,B E21-FS-N006A,B E41-FSL-N006	Barton Differential Pressure Switches	(1),(2),(5)
22	NONE	E51-FS-N002	Barton Differential Pressure Switch	(2)
23	100	CAC-TE-1258-1 thru 14 CAC-TE-1258-17 thru 24	Pyco Temperature Elements	(2)
24	107,108 110 thru 112	E41-TS-3314 thru 3318 E41-TS-3354 E41-TS-3488 E41-TS-3489 E51-TS-3319 thru 3323 E51-TS-3355 E51-TS-3487	Fenwal Temperature Switch	(1),(2)
25	109	B21-TS-N010A thru D	Fenwal Temperature Switches	(2),(4)
26	115	1(A-D)-BFIV-RB-L	NAMCO Position Switch	(2)
28	124,125,126 127,128,129	B32-F019-L B32-F020-L CAC-V47-L CAC-V48-L CAC-V55-L CAC-V56-L CAC-PV-1200B CAC-PV-1205E-L1 CAC-PV-1205E-L2 CAC-PV-1209A-L1 CAC-PV-1209A-L2	Honeywell Limit Switches	(2),(4),(5)

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
28 (Cont'd_		CAC-PV-1209B		
		CAC-PV-1211E		
		CAC-PV-1211F-L1		
		CAC-PV-1211F-L2		
		CAC-PV-1215E		
		CAC-PV-1225B		
		CAC-PV-1227A-L1		
		CAC-PV-1227A-L2		
		CAC-PV-1227B-L1		
		CAC-PV-1227B-L2		
		CAC-PV-1227C		
		CAC-PV-1227E-L1		
		CAC-PV-1227E-L2		
		CAC-PV-1231B		
		CAC-PV-1260-L		
		CAC-PV-1261-L		
		CAC-PV-1262-L		
		CAC-PV-3439-L		
		CAC-PV-3440-L		
		B21-F003-L		
		B21-F004-L		
29	130,131,133 134,135	B11-RS	Honeywell Microswitches	(2)
		B11-RS1		
		B21-CS-3327		
		B21-CS-3329		
		B21-CS-3345		
		B21-CS-3412		
		B41-RS		
		B41-RS1		
		B43-RS		
		B43-RS1		
		B45-RS		
		B45-RS1		
		B46-RS		
		B46-RS1		
		B47-RS		
		B47-RS1		
		B49-RS		
		B49-RS1		
		B50-RS		
		B50-RS1		
		DE3-RS-CS		
		DE3-RS-SS		
		DH2-RS		
		DH2-RS1		
		DH3-RS		
		DH3-RS1		
		DK8-RS		
		DK8-RS1		

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
29 (Cont'd)		DK9-RS		
		DK9-RS1		
		DL0-RS		
		DL0-RS1		
		DL1-RS		
		DL1-RS1		
		DL2-RS		
		DL2-RS1		
		DL7-RS		
		DL7-RS1		
		DL8-RS		
		DL8-RS1		
		DL9-RS		
		DL9-RS1		
		DM1-RS		
		DM1-RS1		
		DM2-RS		
		DM2-RS1		
		DM4-RS		
		DM4-RS1		
		DM5-RS		
		DM5-RS1		
		DM7-RS		
		DM7-RS1		
		DM8-RS		
		DM8-RS1		
		DN1-RS		
		DN1-RS1		
		DN6-RS		
		DN6-RS1		
		DN9-RS		
		DN9-RS1		
		DP2-RS		
		DP2-RS1		
		DP3-CS		
		DP5-RS-A-SS		
		DP5-RS-CS		
		DS4-RS		
		DS4-RS1		
		FN6-CS		
30	132,142,144	MCC-1XA	General Electric IC7700	(2)
	145,146,147	MCC-1XA-2	Motor Control Center	
		MCC-1XB		
		MCC-1XB-2		
		MCC-1XC		
		MCC-1XD		
		MCC-1XDA		
		MCC-1XDB		
		MCC-1XE		

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
30 (Cont'd)		MCC-1XF MCC-1XG MCC-1XH B11-B09-RS GM8-3-6A GN2-3-5A		
31	138	E11-C001A thru D	General Electric Motors	(2)
32	141,155	E41-C002 J16-TB-B thur D	Terry Steam Turbine HPCI Pump System	(1)
33	143	DB0-74-18	Agastat Time Delay Relay	(1)
34	148	D12-RE-N010A,B	General Electric Radiation Detectors	(1),(2),(3),(4)
36	156	NG7-SGT-FILT-1A-RB NG8-SGT-FILT-1B-RB	Farr Standby Gas Treatment System Components	(1),(5)
40	179,181	Terminal Blocks	General Electric Terminal Blocks	(2)
41	182	Terminal Blocks	Curtis Type "L" Terminal Blocks	(2)
42	NONE	C11-F010-L E51-C002-LS4	Namco Limit Switch	(2),(4)
43	NONE	NP6-MOT-M1,M2 NP7-MOT-M1,M2 1A-RX, 1B-RX	DOERR Motors and Control Panels	(3)
44	NONE	E51-C002-H	Square D Float Switch	(2)
45	NONE	B32-CS-F019 B32-CS-F020	Sentry Control Switch	(2)
46	NONE	B21-FT-4157 thru 4167 I2(E-N)-SPLICE I2P-SPLICE	NDT International Accelerometers	(1)(2)

JCO No.	NRC Ter No.	Brunswick Tag No.	Description	10CFR Criteria
48	NONE	QC4-P10-A,B QC4-P10-MA,MB QC4-P2-A,B QC4-P2-MA,MB QC4-P3-A,B QC4-P3-MA,MB QC4-P4-A,B QC4-P4-MA,MB QC4-P5-A,B QC4-P5-MA,MB QC4-P9-A,B QC4-P9-MA,MB QC7-P10-A,B QC7-P10-MA,MB QC7-P6-A,B QC7-P6-MA,MB QC7-P7-A,B QC7-P7-MA,MB QC7-P8-A,B QC7-P8-MA,MB QC7-P9-A,B QC7-P9-MA,MB	Gulton Connector Receptacles and Plugs	(2)
49	NONE	1-IG7-TP5 thru TP8	HPCI System Test Points	(4)
50	NONE	XS4-DS5 XS4-DS6 XS4-MAN.OVRD.SW. XT1-DS11 XT1-DS12 XT1-MAN.OVRD.SW. XT0-DS10 XT0-DS9 XT0-MAN.OVRD.SW. X02-DS7 X02-DS8 X02-MA7.OVRD.SW.	Standby Gas Treatment System Components	(4)

TER NO.: 17

COMPONENT I.D. NO.: E51-F019

MFG/MOD. NO.: LIMITORQUE MODEL SMB-000 VALVE OPERATOR

LOCATION: REACTOR BUILDING -17'

TECHNICAL DISCUSSION:

Component materials of the Limitorque Motorized Valve Operator have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the Class B motor insulation system, melamine switches, and internal wire insulation materials are insensitive to thermal aging effects at the maximum reactor building temperature of 104°F. The valve operators and motor nonmetallic materials are exposed to the plant postulated accident profile which shows a peak temperature of 288°F for 70 seconds, and then drops to 205°F after 100 seconds.

The valve operator is fully qualified for 40 years at the normal and accident reactor building parameters (Reference: Limitorque Test Report No. 600376A).

The Class B motor insulation system has been successfully tested at 250°F for 22.5 hours (Reference: Limitorque Test Report No. B0003). A comparative analysis of the Limitorque "Superheat" test reveals that the internal temperature of the valve operator and motor will not reach 250°F during the initial 100 seconds of accident exposure. Thus, it is judged that the test temperature profile was actually more severe that the plant requirement.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 3

TER NO.: 20

COMPONENT I.D. NO.: B21-F016, E11-F022, E51-F007, G31-F001

MFG/MOD. NO.: LIMITORQUE MODEL SMB-00 VALVE OPERATOR

LOCATION: DRYWELL ELEVATION 17', 80', 50'

TECHNICAL DISCUSSION:

Component materials of the Limitorque Motorized Valve Operators have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the Class H motor insulation system, malamine switches, and internal wire insulation materials are insensitive to thermal aging effects at the maximum drywell temperature of 150°F. The valve operator and motor nonmetallic materials are exposed to the plant postulated accident profile which shows a peak temperature of 298°F.

The valve operators are qualified for 40 years at the normal and accident drywell parameters (Reference: Limitorque Test Report No. 600376A).

The motor, with Class H insulation, has been successfully tested to a peak temperature of 340°F (Reference: Franklin Report No FC-3441) which exceeds the postulated plant accident at BSEP. Additionally, the Class H insulated motors were successfully tested to 2 X 10^8 rads gamma total integrated dose (Reference: Limitorque Report No. FC-3327) which envelops the BSEP requirement of 1.1 X 10^8 Rads gamma.

Thus, it is judged that the Class H insulated motors meet the criteria set forth in 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 24, 25, 26, 27, 28, 29, 30, 35, 36, 37, 38, 39, 41, 42, 44, 45, 47, 48, 49, 50, 52, 53

COMPONENT I.D. NO.:

B32-F020-S	CAC-V47-IS
CAC-PV-1260-S	CAC-V47-S
CAC-PV-1261-S	CAC-V48-IS
CAC-PV-1262-S	CAC-V48-S
CAC-PV-3439-IS*	CAC-V49-S
CAC-PV-3439-S*	CAC-V50-S
CAC-PV-3440-IS*	CAC-V55-S
CAC-PV-3440-S*	CAC-V56-S
CAC-SV-4222	C11-F110A
CAC-SV-4223	C11-F110B
CAC-V4-S	E11-F053A
CAC-V5-IS	E11-F053B
CAC-V5-S	G16-F003 -S
CAC-V6-IS	G16-F004 -S
CAC-V6-S	G16-F019-S
CAC-V7-S	G16-F020-S
CAC-V8-S	RIP-SV-1200A1-1233B2 (150 VALVES)
CAC-V9-S	1A-BFIV-RB-S
CAC-V10-S	1B-BFIV-RB-S
CAC-V15-S	1C-BFIV-RB-S
	1D-BFIV-RB-S

* NO TER

MFG/MOD. NO.:

HB8302C25RU	JV-182-084	HT8316
HT8211B33	HT8262C71	8302
HT8321A6	WPHT8321A1	8321A6
HV-180-414	H8342A4	8262D23

The "HT" AND "HB" prefixes denote high temperature coils with class "H" insulation and are rated for continuous use at 165°F ambient temperature. additionally, documentation for the model 8302 indicated a class "H" was supplied

LOCATION: RHR ROOM, CORE SPRAY ROOM, AND REACTOR BUILDING

TECHNICAL DISCUSSION:

Component materials of the ASCO solenoid valves have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that all the nonmetallic materials, except Buna-N, have greater than 660 years expected life at the maximum 104°F temperature. The Buna-N has an expected life of 11.86 years.

In a letter dated 8-3-79, ASCO stated the following about model numbers HV180-414 and JV182-084:

> "The materials used in the construction of these valves are brass bodies, zinc plate steel bonnets, Buna-N (Nitrile) elastomers, copper shading coils, and all additional internal components are 302, 17-7PH, 305, 416, 430F stainless steel and monel. The valves have Class "H" coils and Nema Type 4 solenoid enclosures.
>
> Based on Engineering judgement, test of similar valves, experience, and rubber manufacturer's literature, the elastomeric components utilized in these valve will function satisfactorily under the accident and post-accident conditions specified in the UE&C Specification. The Class 'H' coils utilized in these valves have been designed for satisfactory operation at 165°F ambient.
>
> Valves of similar design utilizing the said Class 'H' coil system and ethylene propylene elastomers have been satisfactorily qualification tested for use inside containment in accordance with the requirements of IEEE 323-1974, 383-1972, and 344-1975. Part of this test program was a thermal aging test during which the valves were exposed to an ambient temperature voltage and de-energized for 5 minutes every 6 hours. At the completion of this test, the valves functioned satisfactorily with no internal or external leakage. The results of this testing are recorded in ASCO test report AQS21678/TR. Ethylene propylene was chosen as the elastomer in these valves because they are for use inside containment and it is expected that during an accident the temperature could rise to a maximum of 346°F. Since the coils passed the 12 day exposure at 268°F, and rubber manufacturer's literature recommends Buna-N for use at 200°F continuous, it is our opinion that this is justification for stating that these valves are capable of satisfactory operation during the accident and post-accident conditions stated in the UE&C Specification".

Although ethylene propylene was the elastomer in the tested valves, the actual service condition of total time above 200°F of less that 3-minutes followed by a rapid drop off to approximately 135°F for these solenoid at Brunswick is such that Buna-N is an acceptable material.

There is also a Rockwell test report (2972-03-02, Rev. 1; dated 12-1-70) which shows that the HTX8320A20 had successfully functioned during and after exposure to 345° and 110 psig steam for about 2-1/2 hours.

Additionally, a Masoneilan test report (1003, dated 4-19-73) shows that WPHT8300B61 valves successfully functioned during and after exposure to 310°F and 65 psig steam for 23 hours.

Information on radiation damage values shows that the postulated TID of 1 X 10^7 will not significantly degrade the function of the nonmetallic materials except for the acetal disc holder. Testing has been performed on acetal retaining washers to 1 X 10^7 rads with successful results (Reference: MCC Powers Report No. 734-79.002, Rev. 3).

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 6

TER NO.: 34, 113, 114, 123

COMPONENT I.D. NO.: VA-TS-936A, B, C, D, E, F
VA-ZS-936B, A
VA-SV-936B, A

MFG/MOD. NO.: JOHNSON SERVICES; ALLEN BRADLEY

LOCATION: RHR ROOM

TECHNICAL DISCUSSION:

The operation of the RHR Pump Room Cooling Systems has been reviewed. In the event of room A fan cooling unit failure, the room B fan cooling unit will supply the post-LOCA cooling requirements of both RHR pump rooms and the HPCI room simulataneously via interconnecting HVAC ductwork.

The room B fan cooling unit equipment (VA-TS-936B, C, F; VA-ZS-936B; VA-SV-936B) is currently being replaced with fully qualified equipment. This completes the qualification of this redundant backup system.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1) and (i)(5).

Therefore, continued operation is justified.

TER NO.: 62

COMPONENT I.D. NO.: E41-PS-N010
E51-PSL-N006 (NO TER)

MFG/MOD. NO.: STATIC O RING PRESSURE SWITCH 6N-AA21X9SVTT AND 6N-AA21-X9-ST

LOCATION: REACTOR BUILDING EL. -17'

TECHNICAL DISCUSSION:

Component materials of the Static-O-Ring (SOR) pressure switch have been identified and qualification documentation on a similar SOR pressure switch has been obtained. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the lowest expected life of any nonmetallic material used in the pressure switch is 11.86 years.

The pressure switch nonmetallic materials are exposed to the plant postulated accident temperature peak of 288°F for 70 seconds. The accident temperature then decreases to 205°F at 100 seconds and returns to ambient after approximately 20 minutes. This postulated peak temperature transient has been compared to accident test data obtained (212°F for 6 hours) for this switch. Though the testing does not envelop the postulated peak accident temperature, it is judged that no significant detrimental effects to switch operation should occur as a result of the peak temperature transient. This assessment is based on the severity of the test performed in comparison to the short duration of the temperature transient (Reference: Viking Lab Report No. 30203-2).

Additionally, a radiation analysis was performed to determine the threshold of each nonmetallic material used in the pressure switch. It was determined that each material has a radiation threshold greater than the maximum postulated total integrated dose of 2 X 10^6 rads gamma.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 10

TER NO.: 67

COMPONENT I.D. NO.: CAC-PT-1257-2

MFG/MOD. NO.: BAILEY KQ12C

LOCATION: RHR ROOM

TECHNICAL DISCUSSION:

The information provided the operator by these transmitters is also provided by an independent, redundant, and fully qualified transmitter (Rosemount). As such the safety function of this equipment can be accomplished by alternative equipment.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1).

Therefore, continued operation is justified.

TER NO.: 68

COMPONENT I.D. NO.: C32-PT-N005A, B

MFG/MOD. NO.: GENERAL ELECTRIC MODEL 551032GKZZ2 PRESSURE TRANSMITTER

LOCATION: REACTOR BUILDING 50'

TECHNICAL DISCUSSION:

Partial qualification documentation has been obtained for a similar pressure transmitter with the same components and of similar application. The data was evaluated per the DOR guidelines.

The pressure transmitter measures the RPV pressure and gives the operator information regarding plant performance.

Testing has been successfully conducted to show that the device will not fail catastrophically under elevated temperature and humidity conditions (Reference: General Electric Document NSE80036). The accident simulation included a peak temperature of 180°F during which time a 6 point calibration functional test was performed. This was estimated to take about 5 minutes. Additionally, a separate test subjected the transmitter to a 68°F to 158°F at 100% RH test. The tests do not envelop the BSEP requirement of 200°F for 40-50 seconds and the subsequent ramp down to 150°F in 8 minutes. However, the accident peak temperature excursion will not cause significant degradation of equipment operation during that period of exposure above the test maximum temperature (Reference: General Electric Report No. 327, File DV145C3007 and General Electric Document No. NSE80036).

Additionally, analysis indicates that the plant radiation requirement of 1 X 10^5 rads gamma is less than the lowest radiation damage threshold of the transmitter components.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 69

COMPONENT I.D. NO.: E11-PDT-N002A, & B

MFG/MOD. NO.: GENERAL ELECTRIC 552032HKZZ2 PRESSURE TRANSMITTER

LOCATION: REACTOR BUILDING RHR ROOM

TECHNICAL DISCUSSION:

These instruments measure the pressure across the RHR heat exchanger and provide a signal to the RHR service water outlet valve to regulate service water pressure so it is always greater than RHR system pressure. This function can be manually overridden if necessary, and the plant can be safely shutdown in the absence of these devices.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1) and (i)(5).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 13

TER NO.: 71, 72, 73, 74, 76, 77, 78, 79, 80, 81, & 99

COMPONENT I.D. NO.:

E11-PS-N016A	RIP-PSL-1231*	RIP-PSL-1220
E11-PS-N016B	RIP-PSL-1232*	RIP-PSL-1221
E11-PS-N016C	E41-PSH-N012A	RIP-PSL-1222
E11-PS-N016D	E41-PSH-N012B	RIP-PSL-1223
E11-PS-N020A	E41-PSH-N012C	RIP-PSL-1225
E11-PS-N020B	E41-PSH-N012D	RIP-PSL-1227
E11-PS-N020C	E41-PSH-N017A	RIP-PSL-1228
E11-PS-N020D	E41-PSH-N017B	RIP-PSL-1229
RIP-PSL-1200	E51-PS-N020	B32-PS-N018A
RIP-PSL-1201	E51-PSH-N009A	B32-PS-N018A-1
RIP-PSL-1203*	E51-PSH-N009B	B32-PS-N018B
RIP-PSL-1205*	E51-PSH-N012A	SW-TSH-1109
RIP-PSL-1208*	E51-PSH-N012B	SW-TSH-1110
RIP-PSL-1209	E51-PSH-N012C	SW-TSH-1111
RIP-PSL-1210	E51-PSH-N012D	SW-TSH-1112
RIP-PSL-1211	RIP-PSL-1218	IA-PSL-3594,3595*
RIP-PSL-1215*	RIP-PSL-1219	E41-PSH-N027
RIP-PSL-1217	RIP-PSL-1226*	RIP-PSL-1206
SW-PS-1175,1176*		RIP-PSL-1212

* NO TER

MFG/MOD. NO.:

BARKSDALE	B2T-M12SS	D2H-M150SS
	D2T-M18SS	D2T-M150SS
	P1H-M340SS	TC9622-1
	T2H-M251S-12	D2T-M80SS

LOCATION: REACTOR BUILDING, RHR ROOM, CORE SPRAY ROOM

TECHNICAL DISCUSSION:

Component materials of the Barksdale switches have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that all materials, except for Buna-N rubber, have greater than 261 years expected life at the maximum reactor building temperature of 104°F. The switch materials are exposed to the plant postulated accident temperature peak of 288°F for only 70 seconds. The accident temperature then decreases to 145°F within one (1) hour of event initiation. This postulated peak temperature transient has been compared to accident test data obtained (212°F for 6 hours, Ref. AETL TR #596-0398) for those switches. Although the testing does not envelop the postulated peak accident temperature, it is judged that no detrimental effects to switch operation should occur as a result of the peak temperature transient. This is based on the severity of the test performed and the short period of switch exposure to the accident peak temperature.

In addition, the Brunswick switches are located in NEMA 3, 4, 12, or 13 enclosures where the effects of direct steam impingement/humidity would be reduced to nil during the postulated accident.

Also, the component nonmetallic materials have been sucessfully radiation aged during qualification testing (while being used in similar applications) to levels greater than 1 X 10^7 rads gamma, the postulated accident TID for BSEP.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 15

TER NO.: 82

COMPONENT I.D. NO.: E41-LSH-N015A, B

MFG/MOD. NO.: ROBERTSHAW MODEL SL-205-A2-R11-B11-1 LEVEL SWITCH

LOCATION: REACTOR BUILDING -17'

TECHNICAL DISCUSSION:

Partial qualification documentation has been located for the Robertshaw level switches. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques.

The switch nonmetallic components are exposed to the the reactor building postulated accident temperature peak of 288°F for only 70 seconds. In addition, the Brunswick switches are located in a Nema Type 7.9 explosion proof enclosure where the effects of direct steam impingment/humidity would be reduced to nil during the postulated accident. The accident temperature requirement then decreases to 145°F within one (1) hour of event initiation. This postulated peak temperature transient has been evaluated and compared to the accident test data obtained (212°F, 10 psig for 5 hours, Reference: Robertshaw unnumbered test report dated March 28, 1983) for these switches.

Although the testing does not envelop the postulated peak accident temperature, it is judged that no detrimental effects to switch operation will occur as a result of the peak temperature transient. This is based on the severity of the test performed and the short exposure time oF the level switches to the 288°F accident peak.

Also, the component nonmetallic materials have been successfully radiation aged during qualification testing (while being used in similar applications) to levels greater than the BSEP requirement of 1×10^7 rads gamma.

Operationally, the level switches located outside containment are used to signal high suppression pool level to the HPCI system.

In the event of a large break LOCA for which the HPCI system cannot maintain RPV level, the switch may be subject to high radiation. However, in this case the HPCI system is not required since the RPV will be depressurized by the break and/or actuation of the ADS system. Adequate core cooling is then provided by the low pressure ECCS systems and safe shutdown does not depend on the operation of this device.

In the event of a small break LOCA for which the HPCI system can maintain RPV level, the core never uncovers and hence core coooling is maintained and the radiation environment is not present. The switch will perform its function prior to an environmentally caused failure since the peak temperature reaches only 145°F.

The 288° environment in this area of the reactor building is due to the HELB event. The function of these switches is to transfer the HPCI suction from the condensate storage tank to the suppression pool on a high suppression pool level condition. Since neither the HELB nor the actions required to mitigate an HELB will result in a high suppression pool level and HPCI system operation at the same time, this function is not needed to mitigate an HELB.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 91

COMPONENT I.D. NO.:

B21-FS-F015A	B21-FS-F015N	B21-FS-F051A	E41-FS-F024C
B21-FS-F015B	B21-FS-F015P	B21-FS-F051B	E41-FS-F024D
B21-FS-F015C	B21-FS-F015R	B21-FS-F057	E51-FS-F044A
B21-FS-F015D	B21-FS-F015S	B32-FS-F009C	E51-FS-F044B
B21-FS-F015E	B21-FS-F043A	B32-FS-F040B	E51-FS-F044C
B21-FS-F015F	B21-FS-F043B	B32-FS-F040C	E51-FS-F044D
B21-FS-F015G	B21-FS-F045A	B32-FS-F055E	
B21-FS-F015H	B21-FS-F045B	B32-FS-F055F	
B21-FS-F015J	B21-FS-F047A	B32-FS-F057A	
B21-FS-F015K	B21-FS-F047B	B32-FS-F057B	
B21-FS-F015L	B21-FS-F049A	E41-FS-F024A	
B21-FS-F015M	B21-FS-F049B	E41-FS-F024B	

MFG/MOD. NO.: MAGNETROL MODEL F-521 FLOW SWITCH

LOCATION: REACTOR BUILDING (VARIOUS ELEVATIONS)

TECHNICAL DISCUSSION:

Component materials of the Magnetrol flow switch have been identified. These materials have been evaluated per DOR quidelines and by applying Arrhenius techniques. Results of the analysis indicate that the nonmetallic components have greater than 47.6 years of expected life at the maximum reactor building temperature of 104°F.

A flow switch of similar design and materials was tested to conditions more severe than the postulated conditions at BSEP for temperature, pressure and relative humidity (Reference: Barton Reports R1-288A-11 and R3-288A-1).

Additionally, a radiation analysis has been performed on each nonmetallic material used in the flow switch. The analysis indicated that each material has a radiation damage threshold level equal to or greater than the maximum postulated total integrated dose of 1 X 10^5 rads gamma.

In addition, an operational analysis has been performed to determine the effects of failure (misleading information, grounds and spurious operation) of these items in both LOCA and HELB environments. The operational analysis indicates that while the flow switch failures could lead to a loss of some associated safety systems or indication, the loss would occur after they were needed or there are alternate systems available to achieve the same safety functions. Sufficient procedural direction and alternate information is available for the operator to diagnose or respond safely to misleading indications.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 18

TER NO.: 93

COMPONENT I.D. NO.: VA-FT-2577

MFG/MOD. NO.: BAILEY BQ13221

LOCATION: REACTOR BUILDING ELEVATION 50'

TECHNICAL DISCUSSION:

Component materials of the Bailey transmitters have been identified and compared to qualification documentation located for transmitters similar in design, construction, and operation. The qualification data has been evaluated per DOR guidelines and by Arrhenius techniques. Results of this evaluation indicate that these transmitters consist of essentially the same materials and components as Rosemount 1153 transmitters. The Bailey transmitter includes Teflon and Viton o-rings. These o-rings are used as static seals between the flange adapter and process flange (Teflon), the process flange and sensor module (Viton), and the electrical housing and cover (Viton). These materials were evaluated at the normal and peak accident conditions and will not experience significant degradation of performance.

The Rosemount transmitters were tested to parameters which envelop the BSEP reactor building conditions (Reference: Rosemount Reports 3788, 108025, and D8300040). Based on the similarity of the Bailey transmitters to the Rosemount transmitters, the testing levels, and the environment at this location (104°F normal, < 200°F for less than 10 minutes peak accident, 1×10^5 rads TID) use of the Bailey transmitters is justifed.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 19

TER NO: 94, 122

COMPONENT I.D. NO.:

B21-F014A-H	B21-F048A & B	CAC-PV-1218C	E41-PV-1220D
B21-F014J-N	B21-F050A & B	CAC-PV-1219B	E41-PV-1221D
B21-F014P	B21-F056	CAC-PV-1219C	E41-F023A-D
B21-F014R, S	B32-F039B & C	CAC-PV-1225C*	E51-F043A-D
B21-F042A & B	B32-F041C	E11-F037A-D	1A-PV-1201A
B21-F044A & B	B32-F056E & F	E41-PV-1218D	CAC-PV-1209D*
B21-F046A & B	B32-F058A & B	E41-PV-1219D	E11-F043A-D

* NO TER

MFG/MOD. NO.: CHERRY E2360H

LOCATION: REACTOR BUILDING 20' AND 50'; RHR ROOM

TECHNICAL DISCUSSION:

Component materials of the Cherry switch have been identified and qualification documentation on a switch of similar materials and application has been located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicated that the nonmetallic components have greater than 66 years expected life at the maximum reactor building temperature of 104°F.

The Cherry switch nonmetallic materials are exposed to the plant postulated accident temperature peak of 288°F for 70 seconds. The accident temperature then decreases rapidly to 205°F at 100 seconds after accident initiation. This postulated peak temperature transient has been compared to accident test data obtained on a similar switch (212°F, 100% RH for 6 hours). Though the testing does not envelop the postulated peak accident temperature, it is judged that no detrimental effects to switch operation should occur as a result of the peak temperature transient. This assessment is based on the severity of the test performed and the short exposure time at the postulated accident peak temperature.

Additionally, radiation testing on switches of the same material and application supports a qualification level of 3.6×10^6 rads gamma, although the testing does not envelop the postulated total integrated dose of 1×10^7 rads gamma, a radiation threshold analysis shows that the radiation threshold analysis for each material used in switch is greater than 1×10^7 rads gamma except for the Delrin button. For the Delrin button there is testing to support the use of this material in a mechanical application to a radiation level of 1×10^7 rads gamma (Reference: MCC Powers Report No. 734-79,002, Rev. 3).

In addition, an operational analysis has been performed to determine the effects of failure (misleading information, grounds, and spurious operation) of these items in both LOCA and HELB environments. The operational analysis indicated that there is sufficient information available for an operator to diagnose a misleading RIP valve position indication to response in a safe manner.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 20

TER NO.: 95

COMPONENT I.D. NO.: E41-FT-N008

MFG/MOD. NO.: GENERAL ELECTRIC 555111BDAA3PDH FLOW TRANSMITTER

LOCATION: RHR ROOM

TECHNICAL DISCUSSION:

This flow transmitter provides control of the HPCI Turbine Control Valve position to maintain design rated HPCI flow. It also provides the control room with an indication of HPCI pump flow.

Partial qualification test data has been obtained and evaluated for the flow transmitter. Testing has been successfully conducted to show that the device will function under elevated temperature and humidity conditions (Reference: G.E. Document No. NSE80036).

The accident simulation included a peak temperature of 180°F for a time sufficiently long enough to perform a 6 point calibration estimated to take about 5 minutes. Additionally, a separate test subjected the transmitter to a 68°F to 158°F at 100%RH test. The tests do not envelop the BSEP requirement of 199°F (3" RCIC line break) for 30 minutes and the subsequent ramp down to 112°F in 8 minutes. However, the accident peak temperature excursion will not cause significant degradation of equipment operation during that period of exposure above the test maximum temperature (Reference: General Electric Report 327, File DV145C3007 and General Electric Document No. NSE80036).

In addition, an operational analysis was performed to address the effects of the postulated accident radiation environments on the operability requirements of the transmitter.

In the event of a large break LOCA for which the HPCI system cannot maintain RPV level, the transmitter may be subject to high radiation. However in this case, the HPCI system is not required since the RPV will be depressurized by the break and/or actuation of the ADS system. Adequate core cooling is then provided by the low pressure ECCS systems. Therefore, operation of this device is not required for safe shutdown. In the event of a small break LOCA for which the HPCI system can maintain RPV level, the core never uncovers, hence cooling is maintained and the harsh radiation evironment is not present.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 96, 97, 98

COMPONENT I.D NO.: E11-PDIS-N021A,B
E21-FS-N006A,B
E41-FSL-N006

MFG/MOD. NO.: BARTON 289

LOCATION: RHR ROOM, CORE SPRAY ROOM

TECHNICAL DISCUSSION:

These items control the minimum flow valves for the RHR, Core Spray and HPCI pumps. A minimum flow valve is generally installed to prevent a pump from running at its shutoff head for an extended period of time.

If the instrument were to fail, showing low flow, the circuit would act to open the valve. Unplanned opening of the minimum flow valve during injection would divert very little emergency flow from the RPV because of flow restricting orifices in each of the minimum flow lines.

If the instrument were to fail, showing high flow, the circuit would act to shut the valve. During injection the valve would already be shut so there would be no effect. Undesirable, unplanned closing of the valve would only occur as the system was being secured by operator action. The operator can be expected to observe this and manually open the valve.

The plant can be safely shutdown without these instruments.

An additional analysis has been performed to insure that pressure switches will maintain electrical integrity during the postulated accident.

Component materials of the Barton differential pressure switches have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the nonmetallic components have greater than 266 years of expected life at the maximum reactor building temperature of 104°F.

The pressure switch nonmetallic materials are exposed to the plant postulated accident temperature peak of 288°F for 70 seconds. The accident temperature then deceases to 205°F at 100 seconds and returns to ambient after approximately 20 minutes. This postulated peak temperature transient has been compared to accident test data obtained (212°F for 6 hours) for this switch (Reference: AETL Test Report No. 596-0399). Though the testing does not envelop the postulated peak accident temperature, it is judged that no significant detrimental effects to switch operation should occur as a result of the peak temperature transient. This assessment is based on the severity of the test performed and the short time for heat transfer through the heavy metal casing.

Additionally, radiation testing on the subject switches supports a qualification level of 3.6 X 10^6 rads gamma. Though the testing does not envelop the postulated total integrated dose of 1 X 10^7 rads gamma, a radiation threshold analysis shows that the radiation threshold for each material used in the switch is greater than 1 X 10^7 rads gamma. For the Viton o-ring there is testing to support the use of this material in an o-ring application up to radiation level of 2 X 10^7 rads gamma (Reference: ASCO Report No. AQR 67368, Rev. 0, paragraph 4.1.4).

This analysis meets the criteria of 10CFR50.49, paragraphs (i)(1), (i)(2), and (i)(5).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 22

TER NO.: NONE

COMPONENT I.D. NO.: E51-FS-N002

MFG/MOD. NO.: BARTON 289

LOCATION: REACTOR BUILDING RHR ROOM

TECHNICAL DISCUSSION:

Component materials of the Barton differential pressure switches have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the nonmetallic components have greater that 266 years of expected life at the maximum reactor building temperature of 104°F.

The pressure switch nonmetallic materials are exposed to the plant postulated accident temperature peak of 288°F for 70 seconds. The accident temperature then decreases to 205°F at 100 seconds and returns to ambient after approximately 20 minutes. This postulated peak temperature transient has been compared to accident test data obtained (212°F for 6 hours) for this switch. Though the testing does not envelop the postulated peak accident temperature, it is judged that no significant detrimental effects to switch operation should occur as a result of the peak temperature transient. This assessment is based on the severity of the test performed and the short time for heat transfer through the heavy metal casing.

Additionally, radiation testing on the subject switches supports a qualification level of 3.6 X 10^6 rads gamma. Though the testing does not envelop the postulated total integrated dose of 1 X 10^7 rads gamma, a radiation threshold analysis shows that the radiation threshold for each material used in the switch is greater than 1 X 10^7 rads gamma except for the Viton O-Ring. For the Viton O-Ring there is testing to support the use of this material in an o-ring application up to radiation level of 2 X 10^7 rads gamma (Reference: ASCO Report No. AQR 67368, Rev.0, paragraph 4.1.4).

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 100

COMPONENT I.D. NO.: CAC-TE-1258-1 TO 14
CAC-TE-1258-17 TO 24

MFG/MOD. NO.: PYCO 100 OHM PLATINUM RTD

LOCATION: DRYWELL

TECHNICAL DISCUSSION:

These temperature elements monitor drywell air space temperature for recording on a multipoint recorder located in the control room.

Pyco has performed qualification testing on similar RTD enveloping BSEP normal and accident service conditions (Reference: Pyco Qualification Test Report No. 16436-82N, Rev. 5, dated 5/18/84).

The similarity of the installed equipment has been confirmed by Pyco.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 24

TER NO.: 107, 108, 110, 111, & 112

COMPONENT I.D. NO.:	E41-TS-3314	E51-TS-3319
	E41-TS-3315	E51-TS-3320
	E41-TS-3316	E51-TS-3321
	E41-TS-3317	E51-TS-3322
	E41-TS-3318	E51-TS-3323
	E41-TS-3354	E51-TS-3355
	E41-TS-3488	E51-TS-3487
	E41-TS-3489	

MFG/MOD. NO.: FENWAL TEMPERATURE SWITCH 17002-40

LOCATION: REACTOR BUILDING EL. -17' AND ABOVE

TECHNICAL DISCUSSION:

These instruments are temperature sensors which monitor temperatures in areas where the HPCI/RCIC steam line is located and initiate an isolation signal in the event of a steam leak in the HPCI/RCIC steam line.

During a LOCA, these switches must not fail in such a way that produces a spurious steam line leak indication until the plant has been brought to a low pressure condition. If such a spurious signal did isolate the HPCI, the redundant ADS system would remain available. No credit is taken for RCIC during a LOCA.

Fenwal temperature switch, Model No. 17002-40 (modified per Patel Engineers specification), has been qualified by testing to meet or exceed BSEP normal and accident conditions. The tested model was identical to the installed one, except the lead wire insulation in the installed switch is teflon.

Teflon has excellent temperature tolerance and the radiation threshold value is 5 X 10^7 rads for electrical applications (Reference: REIC 21). The maximum accident exposure for these switches is 1 X 10^7 rads gamma over 30 days. In the Fenwal temperature switches the Teflon lead wire is sandwiched between two layers of nonradiation sensitive material which will maintain sufficient insulation resistance for the maximum inservice voltage of 120 volts.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1) and (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 25

TER NO.: 109

COMPONENT I.D. NO.: B21-TS-N010A B21-TS-N010B B21-TS-N010C B21-TS-N010D

MFG/MOD. NO.: FENWAL TEMPERATURE SWITCH 17002-40

LOCATION: REACTOR BUILDING (TUNNEL) EL. 20'

TECHNICAL DISCUSSION:

Fenwal temperature switch, Model No. 17002-40 (modified per Patel Engineer's Specification) has been fully qualified by test which exceeds the BSEP normal and accident service conditions (Reference: Patel Engineer's Qualification Report No. PEI-TR-831200-1). The tested model was identical to the one installed at BSEP except the lead wire insulation was different. The installed switches have teflon insulated lead wires and the tested unit had Rockbestos crosslinked polyethylene insulated lead wires.

Teflon has a high temperature rating and the radiation threshold value is 5 X 10^7 rads for electrical applications. (Reference: REIC 21).

These temperature switches initiate main steam isolation valve closure on a high temperature in the steam line tunnel and will complete their safety function immediately after the accident initiation. Therefore, the temperature switch lead wires will not be significantly degraded by an estimated radiation dose of 1.5 X 10^4 rads before completing their safety function.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2) and (i)(4).

Therefore, continued operation is justified.

TER NO.: 115

COMPONENT I.D. NO.: 1(A-D)-BFIV-RB-L

MFG/MOD. NO.: NAMCO D2400XR

LOCATION: REACTOR BUILDING 80'

TECHNICAL DISCUSSION:

Component materials of the NAMCO 2400XR position switch have been identified. The materials have been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this analysis indicate that all materials, except for Buna-N rubber (used as a binder in the asbestos gasket), have greater than forty (40) years demonstrated qualified life at the maximum reactor building temperature of 104°F. The gasket, which is comprised of 20% Buna-N and 80% asbestos, is judged acceptable for continued operation since the Buna-N is used as a binder and once the gasket is properly installed and left undisturbed, no significant degradation would occur.

The analysis performed on the D2400XR switch is based on testing conducted on NAMCO series SL3 switches (generically similar in materials, construction, and operation). These switches were exposed to a 310°F and 65 psig steam environment (Reference: Masoneilon Test Report 1003, dated 4-19-73) which exceeds the BSEP requirement.

A radiation analysis indicates that the lowest damage threshold for the nonmetallic materials is 8.6×10^5 rads gamma. This damage threshold value envelops the BSEP requirements of 1×10^5 rads gamma.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: 124, 125, 126, 127, 128, 129

COMPONENT I.D. NO.:

B21-F003-L#	CAC-PV-1227C*
B21-F004-L#	CAC-PV-1227E-L1
B32-F019-L	CAC-PV-1227E-L2*
B32-F020-L	CAC-PV-1231B*
CAC-PV-1200B*	CAC-PV-1260-L
CAC-PV-1205E-L1*, L2*	CAC-PV-1261-L
CAC-PV-1209A-L1*, L2*	CAC-PV-1262-L
CAC-PV-1209B	CAC-PV-3439-L*#
CAC-PV-1211E*	CAC-PV-3440-L*#
CAC-PV-1211F-L1*, L2*	CAC-V47-L
CAC-PV-1215E*#	CAC-V48-L
CAC-PV-1225B*#	CAC-V55-L
CAC-PV-1227A-L1*, L2*	CAC-V56-L
CAC-PV-1227B-L1*, L2*	

NO TER

MFG/MOD. NO.: HONEYWELL MODEL OP-AR AND *OPD-AR LIMIT SWITCHES

LOCATION: DW 17' (B32-F019, B21-F003, B21-F004 ONLY)
RX 20' & 50' (ALL OTHERS)

TECHNICAL DISCUSSION:

Component materials of the Honeywell limit switches have been identified and partial qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate the limit switches inside the reactor building will perform their post-accident function prior to failure (Reference: (1) "Nuclear Radiation and Switch Applications," Micro Switch, October 7, 1974, (2) "Humidity Test of the 'W' Lever Type '2' Switches with General Purpose Phenolic, Mica-Filled Case and Cover, Melamine or Valox Plungers." Micro Switch, July 15, 1975, (3) "Evaluation of Asbestos-Free Plastics for 250°F Basic Switch," Micro Switch, February 21, 1979, (4) "Environmental Test," 9993 Barksdale, August 13, 1975).

The analysis for the switches located in the reactor building meet the criteria of 10CFR50.49, paragraph (i)(2).

Limit switch plant ID No. B32-F019 located inside the drywell has been type tested for radiation to 1.3 X 10^8 rads gamma, which envelops the BSEP requirement (Reference: "Nuclear Radiation and Switch Application", Micro Switch, October 7, 1974).

However, the test parameters (Reference: (2), (3), and (4) above) do not envelop the BSEP postulated drywell accident conditions.

This switch provides only valve position indication to the control room for the inboard reactor water sample valve (B32-F019). The reactor water sample valve is normally open and may be closed by the control room operator or in response to an automatic isolation signal.

Failure of limit switch B21-F019 has been anlayzed and may result in (1) loss of valve position indication, (2) loss of control power to the valve solenoid, or (3) both (1) and (2). Loss of control power results in automatic closure of the valve. Since control power is fused, electrical fault of the limit switch would not adversely effect other safety related equipment.

However, the plant can be safely shutdown in the absence of limit switch B21-F019 since the valve fails shut and is required to shut for an automatic isolation signal.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2)(4)(5).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 29

TER NO.: 130, 131, 133, 134, AND 135

COMPONENT I.D. NO.:

B11-RS	B49-RS1	DL2-RS*	DM7-RS1
B11-RS1	B50-RS	DL2-RS1	DM8-RS
B21-CS-3327	B50-RS1	DL7-RS	DM8-RS1
B21-CS-3329	DE3-RS-CS*	DL7-RS1	DN1-RS
B21-CS-3345*	DE3-RS-SS	DL8-RS	DN1-RS1
B21-CS-3412	DH2-RS*	DL8-RS1	DN6-RS
B41-RS	DH2-RS1	DL9-RS	DN6-RS1
B41-RS1	DH3-RS*	DL9-RS1	DN9-RS
B43-RS	DH3-RS1	DM1-RS	DN9-RS1
B43-RS1	DK8-RS	DM1-RS1	DP2-RS
B45-RS	DK8-RS1	DM2-RS	DP2-RS1
B45-RS1	DK9-RS	DM2-RS1	DP3-CS
B46-RS	DK9-RS1	DM4-RS	DP5-RS-A-SS
B46-RS1	DL0-RS	DM4-RS1	DP5-RS-CS
B47-RS	DL0-RS1	DM5-RS	DS4-RS
B47-RS1	DL1-RS	DM5-RS1	DS4-RS1
B49-RS	DL1-RS1	DM7-RS	FN6-CS

* NO TER

PLUS 40 VARIOUS MISC. CONTROL SWITCHES

MFG/MOD. NO.: HONEYWELL MICROSWITCH, TYPES: PTSHA202FB52, PTSHA201, PTKBC2221CC, PTKBC2221, PTSHE201, PTK8C221CC, AND PTKBF3221.

LOCATION: REACTOR BUILDING EL. 20'

TECHNICAL DISCUSSION:

The above control and selector switches are in the remote shutdown system and their function is considered as essential passive.

The PT series switch have been tested at 185°F for 767 hours (more than 30 days) as per Honeywell Micro Switch Qualification Report No. 24407. For radiation the switches have been analyzed as per Honeywell Engineering Report No. LTR 15027-1 to be acceptable to 5×10^6 rad TID. BSEP maximum anticipated radiation is 1×10^5 rads. TID.

Honeywell test conditions envelop the BSEP accident duration of 30 days. However, the peak accident temperature of 200°F for 70 seconds was not enveloped. Since the switches are within enclosures, the switches will not see the peak temperature during the short exposure time because of thermal shielding. Moreover, the BSEP accident temperature will remain at 133°F for the remainder of the 30 day post-accident period. Since the switch was exposed to 135°F for more than 30 days, added confidence in the switch's ability to survive the accident and post-accident period is assured.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 30

TER NO.: 132, 142, 144, 145, 146, & 147

COMPONENT I.D. NO.: MCC-1XA, MCC-1XA-2, MCC-1XB, MCC-1XB-2, MCC-1XC, MCC-1XD, MCC-1XDA, MCC-1XDB, MCC-1XE, MCC-1XF, MCC-1XG, MCC-1XH
RELAYS B11-B09-RS, GM8-3-6A, GN2-3-5A*

* NO TER

MFG/MOD. NO.: GENERAL ELECTRIC IC 7700 MOTOR CONTROL CENTER

LOCATION: REACTOR BUILDING

TECHNICAL DISCUSSION:

Test data applicable to the environmental qualification of the General Electric Series IC 7700 motor control center has been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques.

A preliminary assessment of the test data, performed by General Electric Co., indicated that the test data can be used to demonstrate qualification of the motor control centers to be BSEP normal and postulated accident conditions (Reference - Environmental Qualification Assessment Report - Phase I, GE document number 710-03-025B).

Subsequent to the preliminary assessment, GE issued a second document, GE report number NEDC-30322-P. This document contains detailed Engineering Change Notice (ECN) reviews, Product Analysis Reports, Similarity Analysis Reports on specific components contained in the motor control centers (such as; THED circuit breakers, CR109 magnetic starters, control power transformers, relays and other components). This report also indicates that the best data obtained demonstrated qualification of the IC 7700 motor control center to the BSEP normal and postulated accident conditions.

The final report on the qualification status of the IC 7700 motor control center is currently being prepared by General Electric.

Based upon the test data obtained and the assessments performed, this analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 31

TER NO.: 138

COMPONENT I.D. NO.: E11-C001A, B, C, D

MFG/MOD. NO.: GENERAL ELECTRIC 5K821161C11

LOCATION: REACTOR BUILDING - 50'

TECHNICAL DISCUSSION:

The above motor is a horizontal induction motor with a Class B custom Polyseal insulation. It is a totally enclosed air/water cooled unit designed to operate continuously at 194°F ambient temperature. Its function is to drive the RHR Service Water Booster Pump.

Test data has been obtained for vertical induction motors with the same insulation class (G. E. Document NEDC-30294). The test data obtained envelops the postulted accident conditions at BSEP (temperature, pressure, humidity, radiation).

Arrhenius data obtained for the motor insulation has been evaluated. The evaluation shows a 40 year life for the Class B insulation at the BSEP service conditions.

The motor bearings and lubricating system are inspected and maintained in accordance with the BSEP periodic maintenance and surveillance program.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BESU
JCO NO. 32

TER NO.: 141,155

COMPONENT I.D. NO.: E41-C002 (INCLUDING TERMINAL BLOCKS J16-TB-B, C, D*)

* NO TER

MFG/MOD. NO.: TERRY STEAM TURBINE MODEL CCS HPCI PUMP SYSTEM

LOCATION: REACTOR BUILDING EL. -17'

TECHNICAL DISCUSSION:

An operational analysis has been performed on the Terry Steam Turbine Model CCS HPCI Pump System. The following postulated BSEP accidents were considered in this evaluation:

1. HPCI Steamline Break
2. Large Break LOCA
3. Small Break in RCIC Steamline
4. Small Break LOCA

In all cases alternate qualified ECCS systems in conjuction with the ADS system (auto or manual made) are available to maintain core cooling for a safe shutdown. Operator response is covered in the Emergency Operating Procedures.

This evaluation meets the criterial of 10CFR50.49, paragraph (i) (1).

Therefore, continued operation is justified.

UNIT 1
BESU
JCO NO. 33

TER NO.: 143

COMPONENT I.D. NO.: DBO-74-18

MFG/MOD. NO.: AGASTAT 7022AC TIME DELAY RELAY

LOCATION: REACTOR BUILDING RHR ROOM

TECHNICAL DISCUSSION:

BSEP has one Agastat time delay (model 7022AC) installed in the control circuit of RHR pump room cooler fan A-FCU-RB. An automatic start signal to RHR pump room cooler fan A-FCU-RB de-energizes the coil of the time delay relay which initiates the time delay function. If, after the timer delay setting has elapsed, the fan motor contactor has not closed, an annunciator alarm is sounded in the control room indicating that fan A-FCU-RB has failed to start. It is important to note that this relay does not perform any control function to start or stop the fan; it only gives indication.

The result of the failure of this relay would possibily be: (1) Loss of control power to the fan A-FCU-RB and (2) Loss of alarm to the control room that fan A-FCU-RB has failed to start. If control power is not lost, the fan would start as designed. However, should the first fan fail to start the RHR pump rooms are provided with another 100% capacity fan B-FCU-RB. This fan will automatically start as soon as RHR pump room temperature reaches 145°F or above. There is no time delay relay involved in the control circuit of fan B-FCU-RB.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1)

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 34

TER NO.: 148

COMPONENT I.D. NO.: D12-RE-N010A, B

MFG/MOD. NO.: G. E. MODEL 194 X 927G RADIATION DETECTORS

LOCATION: REACTOR BUILDING EXHAUST AIR PLENUM EL. 80'

TECHNICAL DISCUSSION:

Partial qualification documentation has been obtained for the General Electric radiation detectors. The test data was evaluated per the DOR guidelines and using Arrhenius techniques. The results of this evaluation indicate that the radiation detectors were tested at 212°F for 6 hours and performed satisfactorily before, during and after the test exposure. The test parameters envelop the BSEP requirement of 200°F accident peak temperature (Reference: General Electric Report No. 248A9178).

The reactor building HVAC exhaust air plenum radiation levels are continuously monitored by two redundant radiation detector sensors. The detectors provide output signals which initiate the automatic start of the Standby Gas Treatment System and secondary Containment Isolation when the radiation levels exceed 11 MR/HR.

During normal operation, the total integrated radiation exposure for the detectors will be only 3 X 10^3 rads which is well below the damage threshold level of the detector nonmetallics. The detectors activate at 11 MR/HR and complete its function before damage due to higher levels of radiation is experienced as a result of the accident.

Since the detectors perform their mitigation function immediately upon accident detection, failure would not prevent ECCS actuation or prevent the mitigation of a HELB.

Failure to automatically start the SBGT system and isolate the secondary containment during a HELB will not result in an off-site radiation dose in excess of the 10CFR100 limitations. The resultant radiation release is less than a main steam line break in the turbine building.

SBGT and reactor building isolation may be manually initiated from the control room and/or automatically initiated in response to other sensed parameters which occur during a LOCA.

Additionally, the detectors are periodically tested once every 18 months by physically removing them from their mounting and performing a complete functional test.

This analysis meets the criteria of 10CFR50.49, paragraph (i),(1)(2)(3)(4).

Therefore, continued operation is justified.

TER NO.: 156

COMPONENT I.D. NO.: NG7-SGT-FILT-1A-RB
NG8-SGT-FILT-1B-RB

MFG/MOD. NO.: FARR MODEL NUMBER D51423

LOCATION: REACTOR BUILDING 50'

TECHNICAL DISCUSSION:

The SBGT is not assumed to remain operable in the most severe postulated HELB environment, but as discussed below, its operation is not necessary for this event.

The radioactive release from a HELB in the reactor building is substantially less than that assumed for the main steam line break which is released directly to the atmosphere and results in much less site boundary dose than that permitted by 10CFR100.

Since the inventory loss prior to isolation for a HELB is less than the main steam line break, the offsite HELB dose is also correspondingly low even if the SBGT is not immediately operable. The HELB analyses for BSEP have shown that no fuel damage is expected as a result of the event. Therefore, there will be no excessive radiation levels in the reactor coolant when long term recovery from the event is underway. Thus, there is no need for the SBGT system to maintain a negative pressure in the reactor building during recovery.

This item is located on the 50-foot elevation of the reactor building. The post-LOCA temperature profile in this area is a gradual increase from normal (maximum 104°F) to equilibrium at 133°F in approximately 100 hours. The total integrated radiation dose is 10^5 rads for the 40 year life plus the accident.

Qualification documentation was obtained for the SBGT system and analyzed per DOR Guidelines. The testing was performed on identical and/or similar components (Reference: Farr Test Report No. L-71167). For those safety-related components not tested specifically by The Farr Company, supplemental qualification data was obtained and analyzed. These components include:

1. Blower Motor

   This is an enclosed General Electric blower motor with a Class F insulation system. This insulation system has been analyzed and found to be superior to the G.E. Class B insulation system which has been successfully tested to a 12 hour, 212°F peak temperature, 100% relative humidity and 5.5×10^6 rads gamma. This testing envelops the BSEP postulated accident transient and through analysis, the post-accident period.

2. ITE Molded Case Circuit Breaker

These breakers have been tested separately by ITE at a temperature and radiation dose more severe than the BSEP postulated accident conditions (Reference: ITE-Gould Report No. CC 323.74-57, Rev. 2 dated October 6, 1980).

3. Allen-Bradley Push Button Control and Selector Switches

These devices are manufactured basically from phenolic and metallic materials. Similar switches have been tested by Honeywell to parameters which envelop the BSEP postulated accident conditions (Reference: Honeywell Test Report No. LTR-24407).

4. Allen-Bradley Series 700 Contactor

These contactors have been successfully tested to 2 x 10^8 rads gamma and 248°F which envelops the BSEP requirements (Reference: ANCO letter for IEEE 323-1974 Qualified Components).

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1) and (i)(5).

Therefore, continued operation is justified.

TER NO.: 179, 181

COMPONENT I.D. NO.: TERMINAL BLOCKS

MFG/MOD. NO.: GENERAL ELECTRIC EB-25, CR-151, EB5*

LOCATION: REACTOR BUILDING

* NO TER

TECHNICAL DISCUSSION:

Component materials of the General Electric terminal blocks have been identified and qualification documentation on similar terminal blocks has been located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicates that the nonmetallic components have greater than 5 X 10^8 years of expected life at the maximum reactor building temperature of 104°F.

The test data shows that similar terminal blocks were exposed to test conditions, including radiation, significantly more severe than the postulated accident conditions at BSEP.

Leakage current was monitored during that portion of the test program with conditions at BSEP. The average leakage current per terminal block was less than 1 ma at 120VAC. The results of this test coupled with the facts that:

1. All terminal blocks are in an enclosure and therefore not subjected to direct impingement of steam or water.

2. There is a redundancy of all safety related systems as well as a physical separation.

3. All systems are periodically tested which would detect any random failure.

further substantiate the use of these terminal blocks in the Reactor Building (Reference: Amerace Report F-C5143).

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 41

TER NO.: - 182 -

COMPONENT I.D. NO.: TERMINAL BLOCKS

MFG/MOD. NO.: CURTIS TYPE "L"

LOCATION: DRYWELL/REACTOR BUILDING*

* NO TER

TECHNICAL DISCUSSION:

Test documentation has been located and evaluated for these terminal blocks. A Westinghouse Report PEN-TR-77-83 dated 9/13/77, "Test Report on the Effect of a LOCA on the Electrical Performance of Four Terminal Blocks", and a Westinghouse Research Memo No. 76-1CC-QUAEQ-M24 entitled, "Radiation Hardness of Terminal Blocks", did result in the success of at least four types of similar terminal blocks; Westinghouse, Curtis, Marathon and Cinch Jones. These blocks are similar in material,construction, contact configuration and electrical characteristics to blocks installed at BSEP.

Additionally, Curtis type "L" terminal blocks were tested by Limitorque as part of their qualification of a motorized valve actuator (Limitoque Report No. B-0119). The environmental conditions seen by these test specimens meet the requirements at BSEP. All terminal blocks are in an enclosure and not subjected to direct steam impingement of steam or water. This configuration is similar to the test configuration.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

TER NO.: NONE

COMPONENT I.D. NO.: C11-F010-L
E51-C002-LS4

MFG/MOD. NO.: NAMCO D1200G LIMIT SWITCH

LOCATION: REACTOR BUILDING 50', RHR ROOM

TECHNICAL DISCUSSION:

Component materials of the Namco D1200G limit switch have been identified and qualification documentation on similar equipment located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicates that the nonmetallic components have greater than 9 X 10^3 years at the maximum reactor building temperature of 104°F except for Buna-N. The Buna-N components have an expected life of greater than 11.8 years.

The test data shows that the switch was exposed to test conditions more severe than the BSEP postulated accident conditions for temperature, pressure, and relative humidity (Reference: Masoneilan International Report No. 1003).

Additionally, a radiation analysis performed on the component materials shows that the radiation threshold Buna-N which is the weakline material 1 X 10^6 rads. The switches scomplet their safety function is less than one hour and the maximum postulated total integrated radiation dose during this time is 1 X 10^5 rads which is much lower that the Buna-N threshold value of 1 X 10^6 rads.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2) and (i)(4).

Therefore, continued operation is justified.

TER NO.: NONE

COMPONENT I.D. NO.: NP6-MOT-M1, M2 1B-RX   NP7-MOT-M1, M2 1A-RX

MFG/MOD. NO.: DOERR MOTORS AND ITE CONTROL PANELS

LOCATION: REACTOR BUILDING EL. 20'

TECHNICAL DISCUSSION:

The above electrical components are associated with the compressors to the standby air supply for the Non-Interruptible Air System. Non-interruptible instrument air is supplied to the following control systems:

1. Main steam isolation valves
2. Scram valves
3. Scram volume vent and drain valves
4. Safety relief valves
5. Control rod drive flow regulators
6. Reactor instrument penetration system valves

Each of the above valves are supplied with air accumulators of sufficient size to provide valve actuation air in the event of total instrument air supply failure. The Control Rod Drive System will perform its required safety function before the compressors will fail as a result of a HELB or LOCA.

A loss of the emergency air compressors could cause a loss of reactor level, pressure and monitoring instrumentation during a LOCA. It could cause a loss of HPCI/RCIC and reactor instrumentation during a HELB until Unit 1's air system could be cross-connected (<1 hour). Alternate systems, instrumentation, or procedural guidance is provided for directing the operator's response during these events. Other safety related components would either complete their safety function before air supply failure, have suitable accumulators, or fail in the safe direction. The air compressors do not directly control any indications.

The above analysis meets the criteria of 10CFR50.49, paragraph (i)(3).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 44

TER NO.: NONE

COMPONENT I.D. NO.: E51-C002-H

MFG/MOD. NO.: SQUARE D 9038-AG1-54 FLOAT SWITCH

LOCATION: RHR ROOM

TECHNICAL DISCUSSION:

This item is part of the RCIC turbine assembly. It must maintain its electical integrity for 30 minutes during the BSEP postulated accident.

Testing has been successfully performed on a HPCI turbine that ocntained this component (Ref: Wyle Lab/Terry Turbine Report No. 20458, R14-21-80). The testing was performed at 150°F for an undetermined time and radiation testing to 1 X 10^6 rads. During the HELB accident condition, the temperature gradually rises from 104°F and peaks at 198°F in 30 seconds at which time steam leak isolation is completed. The accident radiation dose in the first 30 minutes of the accident will be less than 1 X 10^5 rads.

Since the switch terminations are enclosed in a NEMA metal enclosure it is safe to assume that the switch will maintain its electrical integrity for the required duration.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 45

TER NO.: NONE

COMPONENT I.D. NO.: B32-CS-F019
B32-CS-F020

MFG/MOD. NO.: SENTRY MODEL F3N1R1 SWITCH

LOCATION: REACTOR BUILDING EL. 20'

TECHNICAL DISCUSSION:

The Sentry F3N1R1 switch utilizes a Series 2 Honeywell Microswitch as the internal switching mechanism.

Honeywell Series 2 switches have been tested at 149°F for more than 30 days (Reference: Honeywell Microswitch Test Response No. LTR-24407). This test envelops the BSEP accident duration but does not envelop the 70 second BSEP peak temperature transient of 200°F. A material analysis indicates that the switch will not be significantly degraded by the short exposure to the postulated accident peak.

Additionally, the switch has been tested to 1×10^7 Rads (Reference: Honeywell Report No. LTR-15027-1) which envelops the BSEP requirement of 1×10^5 Rads gamma.

This analysis meets the criteria of 10CFR50.49, paragraph (i) (2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 46

TER NO.: NONE

COMPONENT I.D. NO.:	B21-FT-4167	B21-FT-4163	AND	I2J-SPLICE
	B21-FT-4158	B21-FT-4164	I2E-SPLICE	I2K-SPLICE
	B21-FT-4159	B21-FT-4165	I2F-SPLICE	I2L-SPLICE
	B21-FT-4160	B21-FT-4166	I2G-SPLICE	I2M-SPLICE
	B21-FT-4161	B21-FT-4167	I2H-SPLICE	I2N-SPLICE
	B21-FT-4162		I2I-SPLICE	I2P-SPLICE

MFG/MOD. NO.: NDT INTERNATIONAL 78IN/S ACCELEROMETER

LOCATION: DRYWELL EL. 38'

TECHNICAL DISCUSSION:

NDT International accelerometers, Model No. 78IN/S, are qualified on the basis of similarity with the NDT International accelerometer, Model No. 838-1, (Reference Wyle. Qualification Report No. 45638-1). Model 838-1 was fully qualified to meet or exceed all BSEP service conditions inside the drywell.

Similarity

Model No. 78IN/S and 838-1 are similar. The only difference is in the accelerometer cable interface connections.

Should the interface connections fail, there is a possibility of faulty indication of safety relief valve position in the control room. However, another independent indication system is provided for safety relief valve position indication. This redundant channel signal is temperature dependent. Therefore, safety relief valve position indication would not be lost in the event of accelerometer failure.

This analysis meets the criteria of 10CFR50.49, paragraph (i)(1) and (i)(2).

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 48

TER NO.: NONE

COMPONENT I.D. NO.:

QC4-P10-A	QC4-P3-MB	QC4-P9-MA	QC7-P7-B
QC4-P10-B	QC4-P4-A	QC4-P9-MB	QC7-P7-MA
QC4-P10-MA	QC4-P4-B	QC7-P10-A	QC7-P7-MB
QC4-P10-MB	QC4-P4-MA	QC7-P10-B	QC7-P8-A
QC4-P2-A	QC4-P4-MB	QC7-P10-MA	QC7-P8-B
QC4-P2-B	QC4-P5-A	QC7-P10-MB	QC7-P8-MA
QC4-P2-MA	QC4-P5-B	QC7-P6-A	QC7-P8-MB
QC4-P2-MB	QC4-P5-MA	QC7-P6-B	QC7-P9-A
QC4-P3-A	QC4-P5-MB	QC7-P6-MA	QC7-P9-B
QC4-P3-B	QC4-P9-A	QC7-P6-MB	QC7-P9-MA
QC4-P3-MA	QC4-P9-B	QC7-P7-A	QC7-P9-MB

MFG/MOD NO.: GULTON CONNECTOR MODEL RECEPTACLE 9012 & PLUG 9013

LOCATION: DRYWELL

TECHNICAL DISCUSSION:

These items are coaxial cable connectors installed as part of the safety relief valve monitoring system.

The above connectors are located inside the drywell and the prototype unit of the same has been successfully tested in accordance with the requirement of Wyle Qualification Plan No. 45098-1. The testing enveloped the required normal and accident service conditions. However the tested configuration differed from the BSEP operating configuration in that the tested configuration encased the connector in heat shrinkable tubing. This was done after thermal aging to add mechanical rigidity to the assembly.

Since the BSEP configuration is not subject to movement after installation, this added rigidity is desirable but not necessary. The connector is sealed against the environent by a compressible grommet at the cable entry and compressible metal seal where the connector havles join so the sealing capability of the heat shrink is not required.

This analysis meets the criterial of 10CFR50.49(i)(2).

Continued operation is justified.

UNIT 1
BSEP
JCO NO. 49

TER NO.: NONE

COMPONENT I.D. NO.: HPCI SYSTEM TEST POINTS
1-IG7-TP5 1-IG7-TP7
1-IG7-TP6 1-IG7-TP8

MFG/MOD NO.: N/A

LOCATION: REACTOR BUILDING 50'

TECHNICAL DISCUSSION:

These are phenolic 120V control power test jacks that were installed as extensions to the instrument rack terminal block test points.

These items are located in a gasketed enclosure which would minimize the entrance of water or steam. They are electrical "banana" type test points which, because of their simple construction, have a low likelikhood of environmentally related failure due to a steam leak condition. The failure in question would result in small leakage current between the points or between a point and ground similar to but probably less than that experienced for a 120V control power termination on a terminal block. The effect of such a leakage current would be negligible.

Because of the internal spacing and overall construction of this type of component, the likelihood of a failure during the required operating time for the HPCI system (24 hrs.) is vanishingly small.

These will be deleted from the plant.

This anlaysis meets the crieria of 10CFR50.49(i)(4).

Continued operation is justified.

UNIT 1
BSEP
JCO NO. 50

TER NO.: NONE

COMPONENT I.D. NO.: STANDBY GAS TREATMENT SYSTEM COMPONENTS

XS4-DS5	XTO-DS10
XS4-DS6	XTO-DS9
XS4-MAN.OVRD.SW.	XTO-MAN.OVRD.SW
XT1-DS11	X02-DS7
XT1-DS12	X02-DS8
XT1-MAN.OVRD.SW.	X02-MAN.OVRD.SW.

MFG/MOD NO.: ALLEN-BRADLEY INDICATOR LIGHTS (XS4-DS5, XS4-DS6, XTO-DS10, XTO-DS9, XT1-DS11, XT1-DS12, X02-DS7, X02-DS8)

HONEYWELL 914CE1-3 (XS4, XTO, XT1, X02 MANUAL OVERRIDE SWITCHES)

LOCATION: XTO, XT1 REACTOR BUILDING 117'
XS4, X02 REACTOR BUILDING 80'

TECHNICAL DISCUSSION:

These indicator lights and switches act in the control circuits of the supply and exhaust isolation dampers.

During a LOCA, these dampers will close on a safety signal very early in the accident, prior to the environment becoming harsh. Any loss of the control circuit thereafter will not affect the damper positions, it will remain in its safety position.

As discussed below, these items are not necessary to mitigate the effects of an HELB.

The radioactive release from a worst case HELB in the Reactor Building is substantially less than that assumed for the main steam line break which is released directly to the atmosphere and it results in much less site boundary dose than that permitted by 10CFR100.

Since the inventory loss prior to isolation for an HELB is less than the main steam line break, the offsite HELB dose is also correspondingly low even if these dampers do not operate. The HELB analyses for BSEP have shown that no fuel damage is expected as a result of the event. Therefore, there will be no excessive radiation levels in the reactor coolant when long-term recovery from the event is underway. Thus, there is no need for the reactor building to be isolated during recovery.

This analysis meets the requirements of 10CFR50.49, paragraph (i)(4).

Continued operation is justified.