CP&L

Carolina Power & Light Company

NOV 05 1984

SERIAL: NLS-84-462

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. 2
DOCKET NO. 50-324/LICENSE NO. DPR-62
ENVIRONMENTAL QUALIFICATION OF SAFETY RELATED EQUIPMENT
REVISED JUSTIFICATIONS FOR CONTINUED OPERATION

Dear Mr. Vassallo:

By letter dated September 26, 1984, Carolina Power & Light Company submitted the justifications for continued operation (JCOs) for Brunswick-2 equipment qualification. Several of these JCOs were subsequently revised as a result of conversations with your staff. Attachment 1 contains a resubmittal of the entire Brunswick-2 JCO package with those JCOs which have been revised, labeled as Revision 1. In addition, JCO numbers were added to the JCOs for ease of future referencing.

Should you have further questions concerning this matter, please contact Mr. Richard J. Fasnacht at (919) 836-7318.

Yours very truly,

for S. R. Zimmerman
Manager
Nuclear Licensing Section

MAT/ccc (802MAT)

Attachment

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

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ATTACHMENT 1 TO NLS-84-462

JUSTIFICATIONS FOR CONTINUED OPERATION

BRUNSWICK-2

UNIT 2
BSEP
JCO NO. 1

TER NO.: 1

COMPONENT I.D. NO.: CAC-V22

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 insulation system, melamine switches, and internal wire insulation materials are qualified for 40 years at the reactor building maximum service temperature (104°F) and postulated accident conditions. (Reference: Limitorque Test Report No. B003).

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

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 2
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.

TER NO.: NONE

COMPONENT I.D. NO.:	B21-FT-4157	B21-FT-4163
	B21-FT-4158	B21-FT-4164
	B21-FT-4159	B21-FT-4165
	B21-FT-4160	B21-FT-4166
	B21-FT-4161	B21-FT-4167
	B21-FT-4162	

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

LOCATION: DRYWELL EL. 38'

TECHNICAL DISCUSSION:

NDT Iternational accelerometers, Model No. 78IN/S, are qualified on the basis of similiarity 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.

Similiarity

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

Should the interface connection 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.

TER NO.: 180

COMPONENT I.D NO.: TERMINAL BLOCKS

MFG/MOD. NO.: G. E. EB-5

LOCATION: DRYWELL

TECHNICAL DISCUSSION:

EB-5 terminal blocks are used inside the drywell as terminal points for 120V/250V/480V Class 1E control and power circuits only and no low voltage signal circuits are landed on these blocks. The terminal blocks are mounted in Nema 4 enclosures and are not subject to direct steam or water impingement.

Various industry reports indicate that only low voltage signal circuits might be in jeopardy during a DBA. Limitorque Report No. B0119 supports EB-5 terminal block qualification for the DBA at BSEP. Upon receipt and successful analysis of this report, these terminal blocks will be considered fully qualified for this application.

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

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO NO. 3

TER NO.: 20

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

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

LOCATION: DRYWELL ELEVATION 17', 80'

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.

UNIT 1
BSEP
JCO NO. 4

TER NO.:	22, 23, 31, 33, 43, 49, 51, 54, 55	
COMPONENT I.D. NO.:	B21-F004 TD-SV-3598, 3601 C12-F009A,B	B21-F028A, B, C, D B32-F019 SW-124, 125, 126, 129, 130, 131
MFG/MOD. NO.:	ASCO HB8302C25 HTX8320A70 8344A5 HT832322	HT8302 WPHT8321A1 HT8344A5 HB8342A4
LOCATION:	DRYWELL, RHR ROOM CORE SPRAY ROOM, REACTOR BUILDING	

TECHNICAL DISCUSSION:

These valves have been replaced with fully qualified valves (ASCO NP series), are not in service, or no longer have a safety related function and would not cause significant degradation of any safety function under an accident environment.

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

Therefore, continued operation is justified.

UNIT 1
BSEP
JCO NO. 5

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

COMPONENT I.D. NO.: CAC-V4, 5, 6, 7, 8, 9, 10, 15, 47, 48, 49, 50, 55, 56
CAC-SV-4222, 4223
CAC-PV-1260, 1261, 1262
B32-F020
C12-F110A, B
C12-F009A, B
G16-F003, 04, 19, 20
SW-V136, 139
2(A-D)-BFIV-RB
CAC-PV-3439
CAC-PV-3440

MFG/MOD. NO.:

HT832322	JV-182-084	HT8316
HB8302C25RU	HT8262C71	8302
HT8211B33	WPHT8321A1	8321A6
HT8321A6	HB8342A4	8262023
HV-180-414	HT80033	

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 2
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.

UNIT 2
BSEP
JCO NO. 7

TER NO.: 57 & 59

COMPONENT I.D. NO.: CAC-PDS-4222
CAC-PDS-4223
CAC-PSE-N001A-D

MFG/MOD. NO.: BARTON 288A, 289A

LOCATION: REACTOR BUILDING ELEVATION 20' and 50'

TECHNICAL DISCUSSION:

Test data has been obtained which qualifies the subject switches to the BSEP environmental conditions, including the postulated accident conditions (Reference Barton Engineering Reports R3-288A-1 and R1-288A-11).

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

TER NO.: 60, 61, & 63

COMPONENT I.D. NO.:

C72-PS-N002A	E11-PS-N010D	E11-PS-N019C
C72-PS-N002B	E11-PS-N011A	E11-PS-N019D
C72-PS-N002C	E11-PS-N011B	E21-PS-N008A
C72-PS-N002D	E11-PS-N011C	E21-PS-N008B
E11-PS-N010A	E11-PS-N011D	E21-PS-N009A
E11-PS-N010B	E11-PS-N019A	E21-PS-N009B
E11-PS-N010C	E11-PS-N019B	

MFG/MOD. NO.: STATIC O-RING MODEL 12NAA4-X10TT AND SN-AA3-X9-STT PRESSUE SWITCH

LOCATION: REACTOR BUILDING

TECHNICAL DISCUSSION:

Component materials of the Static O-Ring pressure switch have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. This evaluation qualifies the pressure switches to the postulated normal and accident conditions at BSEP for 40 years (Reference: Viking Lab. Report No. 30203-2, dated November 20, 1973).

These pressure swithes complete their safety function in less than 24 hours after the accident initiation.

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

TER NO.: 62

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

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 2×10^6 rads gamma.

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

Therefore, continued operation is justified.

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 P 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.

TER NO.:	71, 72, 73, 74, 76, 77, 78, 79, 80, 81, & 99		
COMPONENT I.D. NO.:	E11-PS-N016A	E41-PSH-N012A	RIP-PSL-1220
	E11-PS-N016B	E41-PSH-N012B	RIP-PSL-1221
	E11-PS-N016C	E41-PSH-N012C	RIP-PSL-1222
	E11-PS-N016D	E41-PSH-N012D	RIP-PSL-1223
	E11-PS-N020A	E41-PSH-N017A	RIP-PSL-1225
	E11-PS-N020B	E41-PSH-N017B	RIP-PSL-1227
	E11-PS-N020C	E41-PSH-N027	RIP-PSL-1228
	E11-PS-N020D	E51-PS-N020	RIP-PSL-1229
	RIP-PSL-1200	E51-PSH-N009A	B32-PS-N018A
	RIP-PSL-1201	E51-PSH-N009B	B32-PS-N018A-1
	RIP-PSL-1206	E51-PSH-N012A	B32-PS-N018B
	RIP-PSL-1209	E51-PSH-N012B	SW-TSH-1109
	RIP-PSL-1210	E51-PSH-N012C	SW-TSH-1110
	RIP-PSL-1211	E51-PSH-N012D	SW-TSH-1111
	RIP-PSL-1212	RIP-PSL-1218	SW-TSH-1112
	RIP-PSL-1217	RIP-PSL-1219	IA-PSL-3594,3595
	SW-PS-1175,1176		
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 1×10^7 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 2
BSEP
JCO NO. 14

TER NO.: 75

COMPONENT I.D. NO.: E51-PS-N019A, B, C, D

MFG/MOD. NO.: BARKSDALE MODEL P1HM85SSV

LOCATION: REACTOR BUILDING ELEVATION 20'

TECHNICAL DISCUSSION:

Test data has been obtained which qualifies the subject switches to the BSEP environmental conditions, including postulated accident conditions. (Reference: AETL Test Report 596-0398)

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

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.: 85

COMPONENT I.D. NO.: B21-LITS-N026A
B21-LITS-N026B

MFG/MOD. NO. YARWAY 4418 EC

LOCATION: REACTOR BUILDING 50'

TECHNICAL DISCUSSION:

Test data has been obtained which qualifies the subject switch to the BSEP reactor building normal and the postulated accident conditions (Reference: Yarway Report No. 3232-3155 and 5628-3509).

The above items have been removed from the list titled "Items to be Deferred Due to Installation Problems".

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO. NO. 17

TER NO.: 91

COMPONENT I.D. NO.:

B21-FS-F015A	B21-FS-F015N	B21-FS-F051A
B21-FS-F015B	B21-FS-F015P	B21-FS-F051B
B21-FS-F015C	B21-FS-F015R	B21-FS-F055
B21-FS-F015D	B21-FS-F015S	B21-FS-1227F
B21-FS-F015E	B21-FS-F043A	E41-FS-F024A
B21-FS-F015F	B21-FS-F043B	E41-FS-F024B
B21-FS-F015G	B21-FS-F045A	E41-FS-F024C
B21-FS-F015H	B21-FS-F045B	E41-FS-F024D
B21-FS-F015J	B21-FS-F047A	E41-FS-F044A
B21-FS-F015K	B21-FS-F047B	E41-FS-F044B
B21-FS-F015L	B21-FS-F049A	E41-FS-F044C
B21-FS-F015M	B21-FS-F049B	E41-FS-F044D

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 1×10^5 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 2
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 X 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 2
BSEP
JCO NO. 19

TER NO.: 94, 122

COMPONENT I.D. NO.: VARIOUS; CAC-PV-1218C, -1219B, -1219C, -1220C, -1221C
E41-PV-1218D, -1219D, 1220D, 1221D

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 that 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.

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 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 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 vent 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.

UNIT 2
BSEP
JCO NO. 21

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 3.6×10^6 rads gamma. Though 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 for each material used in the switch is greater than 1×10^7 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 2×10^7 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 2
BSEP
JCO NO. 22

TER NO.: 97A

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 3.6×10^6 rads gamma. Though 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 for each material used in the switch is greater than 1×10^7 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 2×10^7 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 2
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 5×10^7 rads for electrical applications (Reference: REIC 21). The maximum accident exposure for these switches is 1×10^7 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 2
BSEP
JCO NO. 25

TER NO.: 109

COMPONENT I.D. NO.: B21-TS-N010A B21-TS-N010C
B21-TS-N010B 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 wir nsulation was different. The installed switches have teflon insulat 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.

UNIT 2
BSEP
JCO NO. 26

TER NO.: 115

COMPONENT I.D. NO.: 2(A-D)-BFIV-RB

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 significaant 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 requiremnt of 1×10^5 rads gamma.

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

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO NO. 27

TER NO.:	116, 117, & 118	
COMPONENT I.D. NO.:	CAC-V9 CAC-V10 CAC-V15	CAC-V49 CAC-V50
MFG/MOD. NO.:	BETTIS TYPE RX-41 AND RX-341	
LOCATION:	REACTOR BUILDING 50' AND 107'	

TECHNICAL DISCUSSION:

Component materials of the Bettis Limit 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 switch mechanism (Microswitch type BZ) have greater than forty (40) years demonstrated qualified life at the maximum reactor building temperature (104°) and postulated accident conditions. (References: (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," July 15, 1975. (3) "Evaluation of Asbestos-Free Plastics for 250° Basic Swith," Micro Switch, February 21, 1979. (4) "Environmental Test," 9993 Barksdale, August 13, 1975.

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO NO. 28

TER NO.:	124, 125, 126, 127, 128, 129	
COMPONENT I.D. NO.:	B32-F019, B32-F020	CAC-PV-1227A*
	CAC-V47, CAC-V48	CAC-PV-1227B*
	CAC-V55, CAC-V56	CAC-PV-1227C*
	CAC-PV-1200B*	CAC-PV-1227E*
	CAC-PV-1205E*	CAC-PV-1231B*
	CAC-PV-1209A*	CAC-PV-1260
	CAC-PV-1209B*	CAC-PV-1261
	CAC-PV-1211E*	CAC-PV-1262
	CAC-PV-1225B*	B21-F003
		B21-F004
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 2
BSEP
JCO NO. 29

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

COMPONENT I.D. NO.:

DL8-RS1	DM7-RS1	DL0-RS1	B43-RS1
DL9-RS1	DM8-RS1	DL1-RS1	DH3-RS1
DM4-RS1	DN6-RS1	DL2-RS1	DH2-RS1
DM5-RS1	DK8-RS1	DS7-RS1	B50-RS1
B11-RS1	B41-RS1	B45-RS1	B49-RS1
B47-RS1	B11-RS	B21-CS-3412	B21-CS-3327
B21-CS-3329		B21-CS-3345	

MFG/MOD. NO.: HONEYWELL MICROSWITCH, TYPES: PTSEA202FB52, TPSHA201, PTKBC2221CCF9, PTKBC2221, AND PTSHE202CB97

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 X 10^6 rad TID. BSEP maximum anticipated radiation is 1 X 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 2
BSEP
JCO NO. 30

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

COMPONENT I.D NO.: MCC-2XA, MCC-2XA-2, MXX-2XB, MCC-2XB-2, MCC-2XC, MCC-2XD, MCC-2XDA, MCC-2XDB, MCC-2XE, MCC-2XF, MCC-2XH, MCC-2XJ, MCC-2XK

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, G. E document number 710-03-025B).

Subsequent to the preliminary assessment, G. E. issued a second document, G. E. report number NEDC-30322-P. This document contains detailed Engineering Change Notice (ECN) reviews, Product Analysis Reports, and Similiarity Analysis Reports on specific components contained in the motor control centers (THED circuit breakers, CR109 magnetic starters, and a control power transformer). This report also indicates that the test 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 2
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.

TER NO.: 141,155

COMPONENT I.D. NO.: E41-C002

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 2
BESU
JCO NO. 33

TER NO.: 143

COMPONENT I.D. NO.: DBO-74-17

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 2
BSEP
JCO NO. 34

TER NO.: 148

COMPONENT I.D. NO.: D12-PE-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 3×10^3 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.: 151

COMPONENT I.D. NO.: RING AND TONGUE TERMINATION LUGS

MFG/MOD. NO: AMP (NYLON INSULATION SLEEVE)

LOCATION: DRYWELL

TECHNICAL DISCUSSION:

The nylon insulated lugs are used to terminate Class 1E cables inside the drywell at the Penetration Termination Boxes. Field inspections were made of these terminals to verify that the lugs were properly aligned and the insulation sleeves were physically separated between adjacent terminals. This spacing is sufficient to prevent shorting of adjacent conductors at the maximum voltage levels without taking credit for the insulating sleeves.

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

Therefore, continued operation is justified.

TER NO.: 156

COMPONENT I.D. NO.: SGT-FILT-2A-RB
SGT-FILT-2B-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.

UNIT 2
BSEP
JCO NO. 37

TER NO.: 164

COMPONENT I.D. NO.: RAYCHEM CONTROL CABLE

MFG/MOD. NO.: RAYCHEM/FLAMTROL CABLE

LOCATION: DRYWELL, REACTOR BUILDING

TECHNICAL DISCUSSION:

The cable discussed in TER 164 is Raychem/Flamtrol, unshielded multiconductor cable rated at 1000 volts having a combined insulation thickness of 120 mils or greater. This cable type was subjected to testing in a program submitted for NRC review in letters dated October 20, 1983 and May 16, 1983. The cable was tested satisfactorily during the period June to July 1984. The final reports for that testing program have not yet been complete.

Upon receipt and successful analysis of this final report, this cable type will be considred fully qualified for this application.

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

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO NO. 38

TER NO.: 169

COMPONENT I.D. NO.: NONE

MFG/MOD. NO.: PYLE NATIONAL MODEL NS2 CONNECTOR

LOCATION: RX 107'

TECHNICAL DISCUSSION:

Component materials of the PYLE National connectors have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. This evaluation qualifies the connectors to the postulated normal and accident conditions at BSEP for 40 years (Reference: PYLE National Report No. TRC-01637-QL).

The above items have been removed from the list titled "Items to be deferred due to qualified replacements not available."

Therefore, continued operation is justified.

TER NO.: 172

COMPONENT I.D. NO.: 5KV TERMINATIONS

MFG/MOD. NO.: BURNDY ELECTRICAL LUGS INSULATED WITH OKONEX BUTYL RUBBER TAPE AND OKONITE NO. 35 JACKETING TAPE

LOCATION: REACTOR BUILDING

TECHNICAL DISCUSSION:

Test data has been located on a similar splice system that justifies the continued use of the 5KV splice system at BSEP (Reference: Okonite Report NQRN-3).

The Burndy electrical lug is an uninsulated, all metal terminal lug used as the 5KV Class 1E cable terminations and is, therefore, insensitive to thermal and radiation degradation.

Of the insulation materials used in the 5KV terminations at BSEP only the Okonex tape was not tested. However, an Arrhenius calculation performed shows an expected life of 330 years at the maximum reactor building temperature of 104°F.

The postulated accident temperature will peak at 288°F 70 seconds after accident initiation, then decline below the U.L. temperature rating of the Okonex at 300 seconds. Although the accident peak exceeds the rating of the material, no significant degradation will occur during the short period of exposure. This is based on time temperature testing of the material which shows that butyl rubber can withstand 100 hours at 290°F prior to significant loss in properties.

Additionally, a radiation analysis performed on the butyl rubber shows that less than 25% loss of elongation occurs after exposure to 1×10^7 rads gamma (Reference: REIC No. 21). This demonstrates minimum degradation at the BSEP requirement.

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

Therefore, continued operation is justified.

UNIT 2
BSEP
JCO NO. 40

TER NO.: 179, 181

COMPONENT I.D. NO.:

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

LOCATION: REACTOR BUILDING - ABOVE 20', RHR ROOM

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 2
BSEP
JCO NO. 41

TER NO.: 182

COMPONENT I.D. NO.: TERMINAL BLOCKS

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

LOCATION: DRYWELL

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.: C12-F010-L
C12-F011-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 is 1 X 10^6 rads. The switches complete 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.

UNIT 2
BSEP
JCO NO. 43

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 2
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 2
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

TER NO.:	NONE	
COMPONENT I.D. NO.:	B21-FT-4157	B21-FT-4163
	B21-FT-4158	B21-FT-4164
	B21-FT-4159	B21-FT-4165
	B21-FT-4160	B21-FT-4166
	B21-FT-4161	B21-FT-4167
	B21-FT-4162	
MFG/MOD. NO.:	NDT INTERNATIONAL 78IN/S ACCELEROMETER	
LOCATION:	DRYWELL EL. 38'	

TECHNICAL DISCUSSION:

NDT Iternational accelerometers, Model No. 78IN/S, are qualified on the basis of similiarity 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.

Similiarity

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

Should the interface connection 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.

TER NO.: 180

COMPONENT I.D NO.: TERMINAL BLOCKS

MFG/MOD. NO.: G. E. EB-5

LOCATION: DRYWELL

TECHNICAL DISCUSSION:

EB-5 terminal blocks are used inside the drywell as terminal points for 120V/250V/480V Class 1E control and power circuits only and no low voltage signal circuits are landed on these blocks. The terminal blocks are mounted in Nema 4 enclosures and are not subject to direct steam or water impingement.

Various industry reports indicate that only low voltage signal circuits might be in jeopardy during a DBA. Limitorque Report No. B0119 supports EB-5 terminal block qualification for the DBA at BSEP. Upon receipt and successful analysis of this report, these terminal blocks will be considered fully qualified for this application.

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

Therefore, continued operation is justified.