CP&L
Carolina Power & Light Company

JUN 2 8 1985

SERIAL: NLS-85-223

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 NOS. 1 AND 2
DOCKET NOS. 50-325 & 50-324/LICENSE NOS. DPR-71 & DPR-62
ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT
JUSTIFICATION FOR CONTINUED OPERATION

Dear Mr. Vassallo:

In meetings held on May 16 and June 6, 1985, Carolina Power & Light Company notified your staff of its intention to request the Commission to grant an extension to the compliance date beyond November 30, 1985 for the Brunswick Unit 2 Environmental Qualification (EQ) program. During the June 6 meeting the Company committed to submit updated Justifications for Continued Operation (JCO) for the staff's information. This submittal is intended to ensure that the staff is fully aware of the current status of the BSEP-2 EQ program.

Attachment 1 summarizes and explains the differences between the current JCOs (Attachment 2) and those submitted on November 5, 1984. These differences include: a tabulation of items previously covered under the term "various"; items which have had their qualification verified since November 5 and therefore no longer require JCOs; and additional items identified since the November submittal as described in Attachment 1. Additionally, wording changes were made to clarify the bases of some of the JCOs.

Should you have any questions or concerns regarding this submittal please contact Mr. Sherwood R. Zimmerman at (919) 836-6242.

Yours very truly,

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

RWS/ccc (1634RWS)

Attachments

cc: Mr. W. H. Ruland (NRC-BNP)
Dr. J. Nelson Grace (NRC-RII)
Mr. M. Grotenhuis (NRC)



A048
1/1

| Number | Nov 5 | Qual'd | Deleted | Delt'd | Added | Total | Notes |
|---|---|---|---|---|---|---|---|
| 1 | 1 | 1 | yes | | | 0 | |
| 2 | 1 | | | | | 1 | |
| 3 | 3 | | | | | 3 | |
| 4 | 16 | 16 | yes | | | 0 | |
| 5 | 36 | 32 | | | | 4 | |
| 6 | 10 | 5 | | | | 5 | |
| 7 | 6 | 6 | yes | | | 0 | |
| 8 | 20 | 20 | yes | | | 0 | |
| 9 | 2 | | | | | 2 | |
| 10 | 1 | | | | | 1 | |
| 11 | 2 | | | | | 2 | |
| 12 | 2 | | | | | 2 | |
| 13 | 51 | 18 | | | | 33 | |
| 14 | 4 | 4 | yes | | | 0 | |
| 15 | 2 | 2 | yes | | | 0 | |
| 16 | 2 | 2 | yes | | | 0 | |
| 17 | 36 | 36 | yes | | | 0 | |
| 18 | 1 | | | | | 1 | |
| 19 | 9 | | | 2 | 53 | 60 | 1 |
| 20 | 1 | | | | | 1 | |
| 21 | 5 | 5 | yes | | | 0 | |
| 22 | 1 | | | | | 1 | |
| 23 | 22 | | | | | 22 | |
| 24 | 15 | 15 | yes | | | 0 | |
| 25 | 4 | 4 | yes | | | 0 | |
| 26 | 4 | | | | | 4 | |
| 27 | 5 | 5 | yes | | | 0 | |
| 28 | 22 | 22 | yes | | | 0 | |
| 29 | 26 | 26 | yes | | | 0 | |
| 30 | 13 | | | 2 | 1 | 12 | 2 |
| 31 | 4 | 4 | yes | | | 0 | |
| 32 | 1 | | | | | 1 | |
| 33 | 1 | | | | | 1 | |
| 34 | 2 | | | | | 2 | |
| 35 | 1 | | | | | 1 | |
| 36 | 2 | | | | | 2 | |
| 37 | 1 | 1 | yes | | | 0 | |
| 38 | 1 | 1 | yes | | | 0 | |
| 39 | 1 | 1 | yes | | | 0 | |
| 40 | 2 | 1 | | | | 1 | |
| 41 | 1 | 1 | yes | | | 0 | |
| 42 | 3 | 1 | | | | 2 | |
| 43 | 6 | | | | | 6 | |
| 44 | 1 | | | | | 1 | |
| 45 | 2 | | | | | 2 | |
| 46 | 11 | | | | | 11 | |
| 47 | 1 | | | | | 1 | |
| 48 | n/a | | Added | | 1 | 1 | 3 |
| 49 | n/a | | Added | | 2 | 2 | 4 |

| | Listed Nov 5 | Qual'd | Item Delt'd | Items Added | Present Total |
|---|---|---|---|---|---|
| TOTALS => | 364 | 229 | 4 | 57 | 188 |
| JCOs => | 47 | 19 | | 2 | 30 |

The following notes apply to the JCO status listing:

1 The 53 items added to this list are covered under the word "various" in our November 5, 1984 submittal and constitute a more specific listing of these devices.

2 A review of the safety-related loads associated with these MCCs determined that no safety-related loads were supplied by MCC-2XJ or MCC-2XK. Therefore, these MCCs were deleted from the JCO. MCC-2XM was found to supply safety-related loads and therefore, included on the JCO.

3 This new item is a JCO for the Amphenol connectors used on the Safety Relief Valve position indication system. These position indicators are listed in JCO 46.

4 These items were added when a like-in-kind replacement was made with similar, but not identical, devices.

ATTACHMENT 2

TO SERIAL NO.: NLS-85-223

BRUNSWICK-2

JUSTIFICATIONS FOR CONTINUED OPERATION

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 than the plant requirement.

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

Therefore, continued operation is justified.

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.

TER NO.: 24, 25, 26

COMPONENT I.D. NO.:

CAC-SV-4222
CAC-SV-4223
CAC-V4-S
CAC-V5-S

MFG/MOD. NO.: HT8321A5
ASCO HT8321A6
8321A6

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:

> "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 of 268°F for 12 days. The valves were continuously energized at nominal 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.

TER NO.: 34, 113, 114, 123

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

MFG/MOD. NO.: SV - JOHNSON SERVICES V24
TS - JOHNSON SERVICES TS-A19AAC9
ZS - ALLEN BRADLEY 802T-A

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 Qualified 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.

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.

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 change in 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.

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

COMPONENT I.D. NO.:

| | | | |
|---|---|---|---|
| E11-PS-N016A | E41-PSH-N012A | E51-PS-N020 | B32-PS-N018A |
| E11-PS-N016B | E41-PSH-N012B | E51-PSH-N009A | B32-PS-N018A-1 |
| E11-PS-N016C | E41-PSH-N012C | E51-PSH-N009B | B32-PS-N018B |
| E11-PS-N016D | E41-PSH-N012D | E51-PSH-N012A | SW-TSH-1109 |
| E11-PS-N020A | E41-PSH-N017A | E51-PSH-N012B | SW-TSH-1110 |
| E11-PS-N020B | E41-PSH-N017B | E51-PSH-N012C | SW-TSH-1111 |
| E11-PS-N020C | | E51-PSH-N012D | SW-TSH-1112 |
| E11-PS-N020D | | | IA-PSL-3594,3595* |
| SW-PS-1175,1176* | | | E41-PSH-N027 |

* 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.

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.

TER NO.: 94, 122

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

and including items identified as various on Rev. 0 which are:

| | | | |
|---|---|---|---|
| B21-F014A-H | B21-F014J-N | B21-F014P | B21-F014R&S |
| 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-1209D* | CAC-PV-1225C* |
| E11-F037A-D | E11-F043A-D | E41-F023A-D | E51-F043A-D |
| IA-PV-1201A | | | |

* 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 295°F for 70 seconds. The accident temperature then decreases rapidly to 205°F at 100 seconds after accident initiation. The recommended maximum service temperature for continued use of these non-metallic materials exceed the peak temperature of 295°F with the exception of DELRIN. This plastic material has a 220°F maximum service temperature but has heat deflection temperature of 316F at 66 psi and, therefore, will not be affected during a short exposure to 295°F.

Since CP&L has successfully qualified a switch of similar configuration (Environmental Qualification Test Report for Electrical Devices. Brunswick Steam Electric Plant, Units 1 & 2, CC&L Project #84-1857) to conditions enveloping the BSEP worst case, it is judged that no detrimental effects to switch operation will 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 $3.6 \times 10^6$ rads gamma, although the testing does not envelop the postulated total integrated dose of $1 \times 10^7$ rads gamma, a radiation threshold analysis shows that the radiation threshold analysis for each material used in switch is greater that $1 \times 10^7$ 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 $1 \times 10^7$ rads gamma (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.

TER NO.: 95

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

MFG/MOD. NO.: GENERAL ELECTRIC 50-555111BDAA3PDH 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.

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 $3.6 \times 10^6$ rads gamma. Though the testing does not envelop the postulated total integrated dose of $1 \times 10^7$ rads gamma, a radiation threshold analysis shows that the radiation threshold for each material used in the switch is greater than $1 \times 10^7$ 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 $2 \times 10^7$ rads gamma (Reference: ASCO Report No. AOR 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.

TER NO.: 115

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

MFG/MOD. NO.: NAMCO D2400X-R

LOCATION: REACTOR BUILDING 80'

TECHNICAL DISCUSSION:

Component materials of the NAMCO D2400X-R 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 D2400X-R 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: Masoneilan 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 $8.6 \times 10^5$ rads gamma. This damage thresnold value envelops the BSEP requiremnt of $1 \times 10^5$ rads gamma.

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

Therefore, continued operation is justified.

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

COMPONENT I.D NO.: MCC-2XA, MCC-2XA-2, MCC-2XB, MCC-2XB-2, MCC-2XC, MCC-2XD, MCC-2XDA, MCC-2XDB, MCC-2XE, MCC-2XF, MCC-2XH, MCC-2XM*

*No TER

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

LOCATION: REACTOR BUILDING

TECHNICAL DISCUSSION:

GE 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 supplemented by analysis can be used to demonstrate qualification of the motor control centers to the BSEP normal and postulated accident conditions. GE has indicated that it will be necessary to close the conduit entries to the BSEP MCC's in order to meet the assumptions of the Report/Analysis; a program to accomplish this work is in progress. These changes will reduce the calculated peak component temperatures to within the tested parameters. At this time the calculated peak component temperatures exceed the GE test parameters by approximately 10°F (approx. 140°F vs. 131°F). Typical components of the type contained within a motor control center would not be adversely affected by this short time increase in temperature, CP&L is in possession of a type test of another manufacturers MCC, with a much higher peak temperature (250°F for 6.2 hrs.) which tends to support this conclusion.

Subsequent to the preliminary assessment, G. E. issued a second document, which 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, subject to the above configuration changes.

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.

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 conjunction with the ADS system (auto or manual mode) are available to maintain core cooling for a safe shutdown. Operator response is covered in the Emergency Operating Procedures.

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

Therefore, continued operation is justified.

TER NO.: 143

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

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

LOCATION: REACTOR BUILDING RHR ROOM

TECHNICAL DISCUSSION:

BSEP has one Agastat time delay relay (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 that is fully qualified. 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.

TER NO.: 148

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

MFG/MOD. NO.: G. E. MODEL 194X927G11 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.: 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, but are not limited to, the following major components.

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 $5.5 \times 10^6$ 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 $2 \times 10^8$ 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), (i)(2) and (i)(5).

Therefore, continued operation is justified.

TER NO.: 179, 181

COMPONENT I.D. NO.: TERMINAL BLOCKS

MFG/MOD. NO.: GENERAL ELECTRIC, 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.

TER NO.: NONE

COMPONENT I.D. NO.: C12-F010-L
E51-C002-LS4

MFG/MOD. NO.: NAMCO D1200G, D-200G-ST-2 LIMIT SWITCHES

LOCATION: REACTOR BUILDING 50', RHR ROOM

TECHNICAL DISCUSSION:

Component materials of the Namco D1200G and D-200G-ST-2 limit switches 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 weaklink material is 1 X $10^6$ rads. The switches complete their safety function in 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.: 2-NP6-MOT-M1, M2; 2A-RX
2-NP7-MOT-M1, M2; 2B-RX

MFG/MOD. NO.: DOERR MOTORS, A421; LINCOLN MOTORS, 213T; AND ITE CONTROL PANELS, CDP-7

LOCATION: REACTOR BUILDING EL. 20'

TECHNICAL DISCUSSION:

The above electrical components are associated with the compressors for the standby air supply for the Non-Interruptible Air System.

Non-Interruptible Air is normally supplied from the plant service air system, these compressors are a standby source to be used upon loss of the normal supply. 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. Reactor instrument penetration (RIP) system valves

Items 1, 2 and 3 above will fail to a safe position on loss of air.

The Safety Relief valves are supplied with air accumulators of sufficient size to provide valve actuation air in the event of total instrument air supply failure.

The RIP valve control system contains provisions to isolate the air-operated valves on loss of air system pressure. This will trap air in the header and operating lines maintaining the RIP isolation valve in its required position, components necessary to perform this function are qualified.

Additionally, normal station service power to the service air system can be restored within a reasonable time (< 1 hour), and BSEP has the capability to cross connect the air systems between the two units which would allow the non-affected unit to supply air for these loads.

The air compressors do not directly control any indications.

The above analysis meets the criteria of 10CFR50.49, paragraph (i)(1), (i)(4) and (i)(5).

Therefore, continued operation is justified.

TER NO.: NONE

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

MFG/MOD. NO.: SQUARE D 9038-AG-1-S4 FLOAT SWITCH

LOCATION: RHR ROOM

TECHNICAL DISCUSSION:

This item was supplied as part of the RCIC turbine assembly, and is used to perform a non-safety-related control function. This component is passive and must not fail so as to jeopardize the isolation logic it is powered from.

Testing has been successfully performed on a HPCI turbine that contained 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 295°F in 60 seconds at which time steam leak isolation is completed. The temperature returns below 150°F in 15 minutes. The accident radiation dose in the first 24 hours of the accident will be less than 1 X $10^6$ 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.

TER NO.: NONE

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

MFG/MOD. NO.: SENTRY MODEL F3N1R1 SWITCH

LOCATION: REACTOR BUILDING EL. 50'

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 $1 \times 10^7$ Rads (Reference: Honeywell Report No. LTR-15027-1) which envelops the BSEP requirement of $1 \times 10^5$ 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<br>B21-FT-4158<br>B21-FT-4159<br>B21-FT-4160<br>B21-FT-4161<br>B21-FT-4162 | B21-FT-4163<br>B21-FT-4164<br>B21-FT-4165<br>B21-FT-4166<br>B21-FT-4167 |
| 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. These connections are vendor supplied splices between a mineral insulated and an elastomer insulated cable, they are covered with Raychem nuclear grade heatshrink which has not been tested in this application, but is otherwise qualified.

Should the interface connections fail, there is a possibility of faulty indication of safety relief valve position in the control room. However, the operator will not be misled because other independent indications of safety relief valve position are provided (vessel pressure, suppression pool temperature, vessel level and suppression pool level). Additionally, the non-qualified SRV tailpipe temperature instrumentation may be available. Therefore, safety relief valve position indication would not be lost in the event of accelerometer failure.

This analysis meets the criteria of 10 CFR 50.49, 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.

TER NO.: NONE

COMPONENT I.D. NO.: CONNECTOR

MFG/MOD. NO.: AMPHENOL COAX CONNECTOR
PLUG 34500
RECEPTABLE 36500

LOCATION: INSIDE/OUTSIDE

TECHNICAL DISCUSSION:

Amphenol coaxial connectors are utilized to carry the signal from sensors for safety relief valve position indication. These connectors are hermetically sealed and are provided with another environmental seal by application of Raychem heat shrink tubing.

Raychem heat shrink tubing is fully qualified by Raychem to BSEP service conditions. The connector's organic material is of copolymer styrene and is rated at 390°F. BSEP maximum temperature is only 340°F during the accident condition.

In the unlikely event of a failure of a connector, there is a possibility of faulty indication of safety relief valve position in the Control Room. However, the operator will not be misled because other independent indications of safety relief valve position are provided (vessel pressure, suppression pool temperature, vessel level and suppression pool level). Additionally, the non-qualified SRV tailpipe temperature instrumentation may be available. Therefore, safety relief valve position indication would not be lost in the event of connector failure.

This analysis meets the criteria of 10 CFR 50.49, paragraph (i)(1) and (i)(2).

Therefore, continued operation is justified.

TER NO. NONE

COMPONENT ID NO.: B21-F003L, B21-F004L

MEG/MOD NO.: NAMCO EA 510-17702 LIMIT SWITCHES

LOCATION: DRYWELL 17'

TECHNICAL DISCUSSION:

Component materials of the NAMCO Limit Switches have been identified. The materials have been evaluated per DOR Guidelines and by applying Arrhenius techniques. Results of this analysis indicate that all materials have an expected life of greater than 5 years at the expected average temperature of 120°F. At the time these switches are scheduled for replacement, they will have been installed less than 3 years.

A functional analysis based on the NAMCO EA 740 shows that the EA 510 will remain functional during the accident. The EA /40 was qualified to the same conditions to which the EA 510 will potentially be exposed. The EA 510, like the EA 740, has a NEMA 4 Enclosure containing a polyester-glass switch assembly.

A radiation analysis based on the EPRI document "Radiation Data for Design/Qualification of Nuclear Plant Equipment" shows that the lowest threshold value of damage for the non-metallic materials is $6 \times 10^7$ rads which exceeds the potential integrated dose (including accident) to which these switches may be exposed prior to their scheduled replacement.

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

Therefore, continued operation is justified.