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May 26, 2009

L-MT-09-045
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket 50-263
Renewed Facility Operating License
License No. DPR-22

Monticello Extended Power Uprate: Response to NRC Electrical Engineering Review Branch (EEEB) Request for Additional Information (RAI) dated March 28, 2009, (TAC No. MD9990)

References: 1. NSPM letter to NRC, License Amendment Request: Extended Power Uprate (L-MT-08-052) dated November 5, 2008 Accession No. ML083230111

2. Email P. Tam (NRC) to G. Salamon, K. Pointer (NSPM) dated March 28, 2009, Monticello – "Draft RAI from Electrical Engineering Branch re: proposed EPU amendment (TAC MD9990)" Accession No. ML090880003

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), requested in Reference 1 an amendment to the Monticello Nuclear Generating Plant (MNGP) Renewed Operating License (OL) and Technical Specifications (TS) to increase the maximum authorized power level from 1775 megawatts thermal (MWt) to 2004 MWt.

On March 28, the U. S. Nuclear Regulatory Commission (NRC) Electrical Engineering Review Branch provided fifteen (15) Requests for Additional Information (RAIs) (see Reference 2). Enclosure 1 provides the NSPM responses.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Minnesota Official.

Summary of Commitments

There are no new commitments contained in this letter and no existing commitments were revised by this letter.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on May 26, 2009.

Timothy J. O'Connor
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

ENCLOSURE 1

NSPM RESPONSE TO EEEB RAIs DATED MARCH 28, 2009

NRC RAI No. 1

In Section 3.4.2, the licensee provided an evaluation of D.G. O'Brien and General Electric (GE) electrical penetrations. The staff requests the licensee to confirm that only D.G. O'Brien and GE electrical penetrations are in its environmental qualification (EQ) program. If the licensee has other types of electrical penetrations in its EQ program, the staff requests the licensee to identify those penetrations and provide a discussion and details on how those penetrations are qualified for EPU conditions.

NSPM Response

The EQ Master list only identifies the original plant GE canister penetrations and the D.G. O'Brien penetrations. There are no other electrical penetrations included in the EQ Program at Monticello.

NRC RAI No. 2

In Section 3.4.2, the licensee stated that the Limitorque actuator MO-2397 is qualified per DOR guidelines and by performing a thermal lag evaluation. The staff requests the licensee to confirm that the qualified temperature of Limitorque actuator MO-2397 (i.e., 329°F) adequately envelops the temperature assumed in the thermal lag evaluation.

NSPM Response

The current environmental qualification analysis of the Class H motor for the MO-2397 actuator utilizes the thermal lag testing documented in Limitorque Report B0027. In the test, the specimen actuator was exposed to superheated steam conditions of 385°F for a combined duration of six and a half (6.5) minutes, followed by 365°F for twelve (12) minutes. The specimen actuator had embedded thermocouples within the motor winding and other electrical components of the actuator. The motor sustained no temperature greater than 315°F throughout the superheat exposure. Calculations within Limitorque Report B0027 indicate a thermal lag effect of at least 17 minutes for the internal actuator electrical component or motor. The revised peak drywell temperature under EPU conditions is 338°F (superheated steam). The operating time for MO-2397 of 1 minute does not change at EPU conditions and remains well below the 17 minute test time such that the current thermal lag basis remains applicable under EPU conditions.

NRC RAI No. 3

In Section 3.4.3, the licensee identified several components that would be submerged due to EPU conditions. The licensee stated that these components are not credited for the postulated breaks and submergence level. The staff requests the licensee to confirm that failure of these components under a submerged condition would have no adverse impact on the safety functions of the associated components and systems.

NSPM Response

The failure of these components under a submerged condition has no adverse impact on the safety functions of the associated components and systems at EPU. For clarification, the only components with the potential for being subjected to higher submergence levels under EPU than at CLTP are:

- MO-2107, RCIC pump discharge inboard isolation valve
- MO-2068, HPCI pump discharge inboard isolation valve
- CV-1478, Instrument air to drywell Isolation
- MO-2374, Main steam line drain outboard

All of these valves are located in the Steam Chase and currently are assumed to be inoperable due to submergence under CLTP conditions. Under EPU conditions, only the submergence level increases, the scope of equipment subject to the submergence does not change. As there is no additional equipment subject to potential submergence under EPU, and the current plant safety analyses with the equipment identified above assumed inoperable remains valid for EPU conditions.

NRC RAI No. 4

In Note 4 of Section 3.4.3, the licensee stated that the Rockbestos cables could be submerged and these cables are qualified for 18 hours in a submerged condition. The staff requests the licensee to provide the basis for the 18 hour criteria for qualification, and also confirm that these cables are not submerged more than 18 hours and that these cables will perform their safety function during and post submergence.

NSPM Response

The Rockbestos cables used in this application are not submerged more than 1 hour and will perform their safety function during and post submergence.

The event for which the predicted worst-case 1.7 foot submergence level in Reactor Building Volume 32 is based upon is the "RWCU-B-30-R0" case, all other HELB events affecting this location have less than 6-inches of submergence. The peak submergence level in Volume 32 is reached at 70 seconds, and the level reduces to 0-inches at 1861 seconds. The cables are therefore predicted to be submerged for less than 1 hour. The qualification of the cables is based on Rockbestos Reports QR-5804/5805. The 18-hour (minimum of QR-5804 testing) soak following the simulated LOCA testing was conducted at room temperature with tap water. The cable testing was compliant with IEEE Standard 383-1974 and included mandrel re-bending.

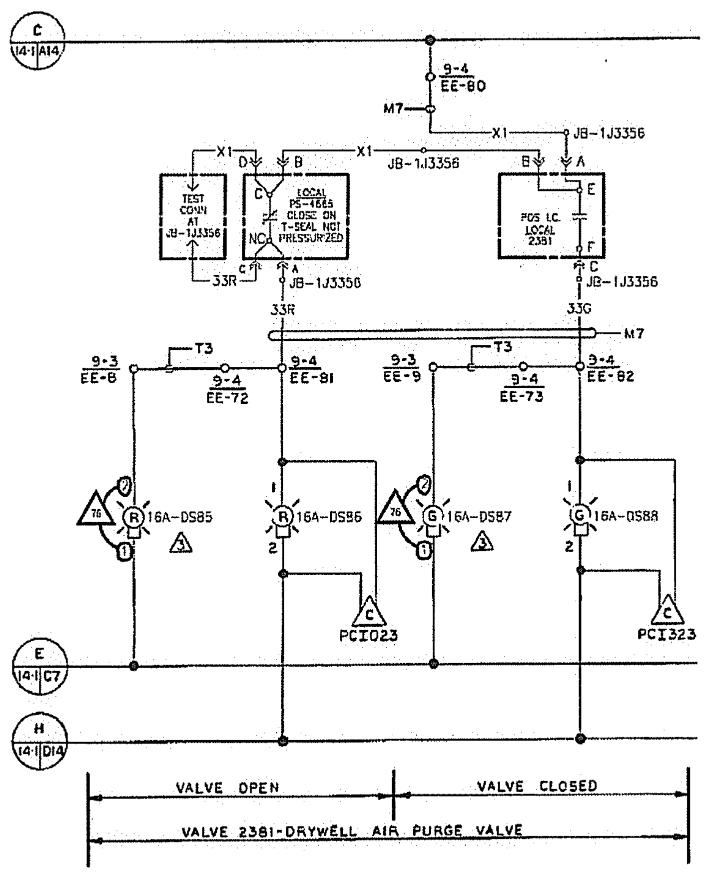
NRC RAI No. 5

In Note 1 of Section 3.4.4, the licensee stated the ASCO pressure switches located in reactor building (RB) Volume 31 serve only as containment isolation valve position indication and are not required under a RB high-energy line break (HELB) event. The staff requests the licensee to confirm that the failure of these switches under EPU conditions would not 1) adversely affect the safety function of the associated components or 2) mislead the operator.

NSPM Response

Reactor Building Volume 31 become harsh due to a RWCU line break. None of the RWCU break detection and mitigation equipment or safe shutdown equipment is affected by the potential failures of the subject pressure switches.

The ASCO pressure switches provide valve position indication for drywell vent valves AO-2386 and AO-2387. See picture below. These valves have no safety function for HELB mitigation, and failure of these switches would not mislead the operator.



AO-2386 and AO-2387 are drywell purge exhaust and vent valves with an elastomer T-ring which requires pneumatic pressure to ensure a leak tight seat. When the valve is fully closed and pressure is on the T-ring, the red position indicating light for the valve will be extinguished by a pressure switch, PS-4666 (AO-2386) and PS-4671 (AO-2387).

The drywell purge exhaust and vent valves are closed during normal power operations. If these pressure switches fail, the valve may indicate both open and close simultaneously. This is of no consequence as the valves have no safety function during the HELB event. This condition would not prevent or change any HELB mitigation actions for this break location from the control room. In this case, the operators would initiate troubleshooting and maintenance as necessary when time allowed.

NRC RAI No. 6

In Note 2 of Section 3.4.4:

NRC RAI No. 6(a)

The licensee stated that Barton Model 580 series pressure switches have no required safety function during a HELB in RB volume 18. The staff requests the licensee to confirm that failure of these switches under EPU conditions would not adversely affect the safety functions of the associated components and systems.

NSPM Response

As indicated in the listed references provided in Note 2 of Section 3.4.4 of T1004, Revision 1, the subject differential pressure switches located in Reactor Building Volume 18 are part of the selection logic for automatic LPCI injection in the event of a design basis LOCA. There would be no breach of the reactor recirculation loops (LOCA) concurrent with the HELB affecting Reactor Building Volume 18. Failure of the LPCI loop select switches DPIS-2-129A/C under the HELB would, therefore, have no impact on ability to inject with LPCI. LPCI injection, if needed, is still available to either recirculation loop.

NRC RAI No. 6(b)

The licensee stated that additional testing from Barton on this switch model would easily demonstrate qualification for the temperature postulated under EPU conditions. The staff requests the licensee to confirm that this test is complete and that the switch is qualified for EPU conditions.

NSPM Response

The additional testing referred to in Task Report 1004, Revision 1 is Barton Report R3-580A-9 which is complete and included as a reference source in the current EQ file. Although the results of this test indicated permanent set point shift in the specimen following the LOCA testing at greater than 340°F, the testing does indicate the survivability of the Model 580 pressure switch when subject to higher temperature conditions. The LPCI loop select pressure switches remain qualified under post-LOCA only operation at EPU power using the current Barton qualification test report R3-580A-29. The independent separate testing documented in Barton Report R3-580A-9 demonstrates that the switches would not severely degrade under the slightly higher temperatures (~210°F) predicted under EPU at the switch location.

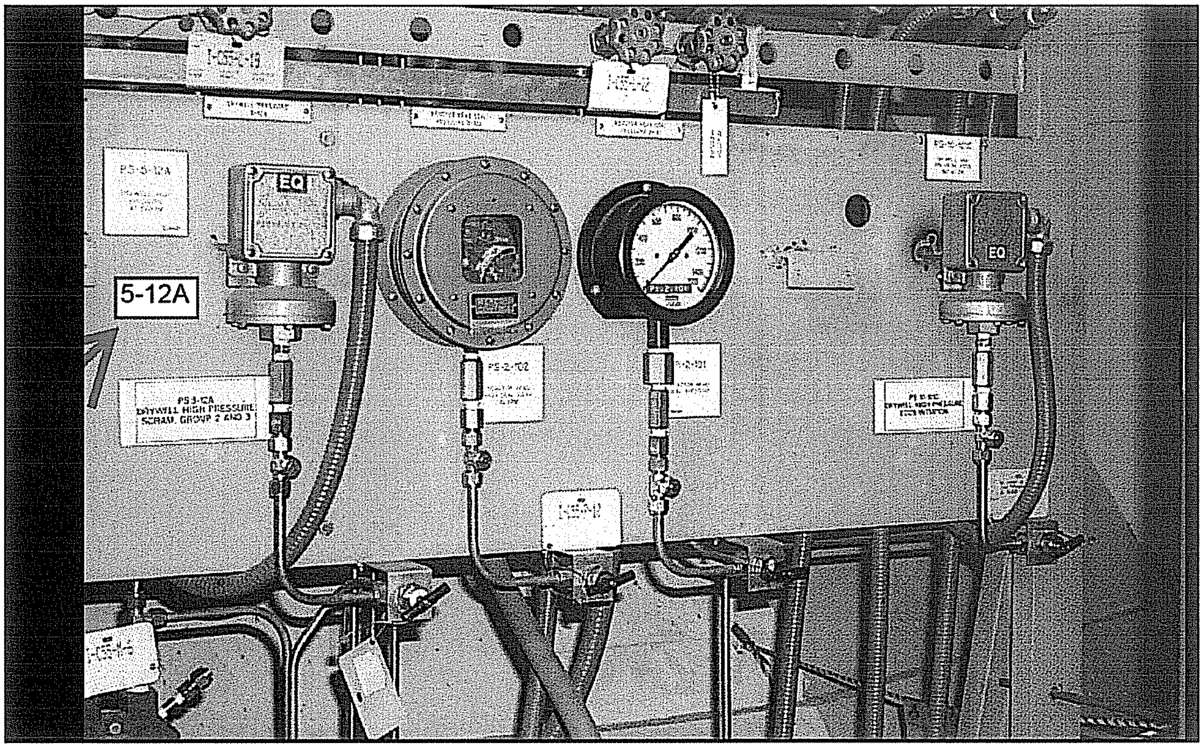
NRC RAI No. 7

In Note 3 of Section 3.4.4, the licensee stated that 1) the ITT Royal cable is only installed in local instrument panels in RB Volumes 14, 18, 19, 22, and 33 and that these cables are qualified for peak temperature of 211°F, 2) the EPU HELB temperatures in RB Volumes 14, 18, 19, 22, and 33 are 174.7°F, 209.7°F, 209.3°F, 184°F, and 211.7°F, respectively, and 3) using engineering judgment that the HELB event would not exceed the test level of 211°F as only a 0.7°F difference between the test level and the predicted values exists. The staff requests the licensee to confirm that these cables are not exposed to any temperature rise or hot spots in the panels. If these cables are exposed to hot spots in the panels, then the staff requests the licensee to provide the qualification basis for these cables.

NSPM Response

The cables are not exposed to any temperature rise or hot spot conditions in the panels. The instrument panels discussed are not enclosed cabinets but rather the original General Electric plant instrument racks on which various pressure transmitters, switches, or other system instrument monitoring devices are located. As indicated by the data presented in Note 3 of Section 3.4.4 of T1004, Revision 1, the ITT Royal cable is qualified with margin (1.3°F minimum) for use in Reactor Building Volumes 14, 18, 19, and 22. The ITT Royal cable has been identified as original plant equipment and is environmentally qualified in accordance with the DOR Guidelines.

The predicted peak EPU HELB temperature in Reactor Building Volume 33 exceeds the test temperature of 211°F by only 0.7°F. The ITT Royal cable is only used on pressure switches PS-5-12A/B on instrument rack C-55 (25-5) in Reactor Building Volume 33. Plant drawing NX-7829-31-1 provides the layout of the instrument rack which shows that only non-heat producing measuring instruments are mounted on the rack (see picture below). The drawing also specifies the use of individual conduit routings to the local junction box for each device. PS-5-12A/B have normally closed contacts and serve to energize relays 5A-K4A and 5A-K4B, respectively. These relays are General Electric type HFA and are the only load on PS-5-12A/B pressure switches (and thus the ITT Royal cable). As such, internal cable heating is insignificant.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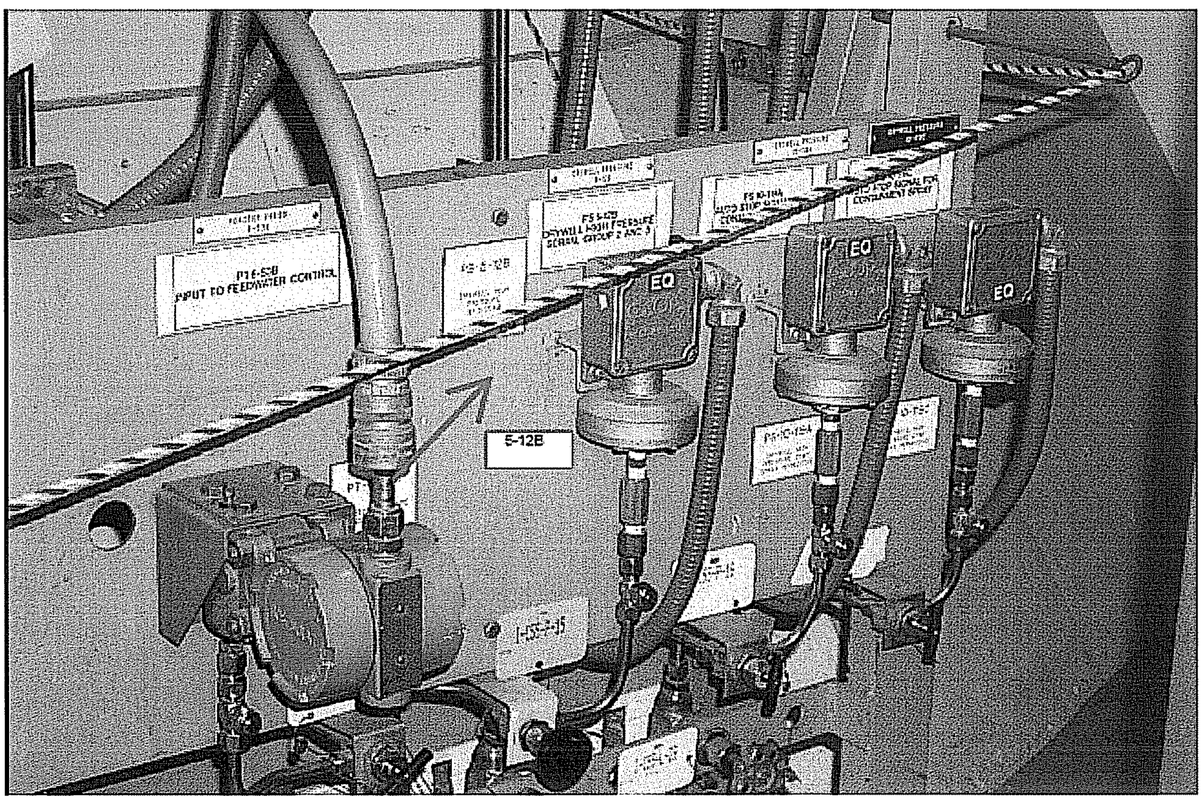
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NRC RAI No. 8

In Section 3.4.5, the licensee stated that the turbine building (TB) HELB review revealed that the Rockbestos Firewall III, Rockbestos EP, and Brand Rex cables are included in the EQ program. The licensee further stated that these cables are similar types to those already qualified for bounding conditions present in either the drywell or the RB. The licensee concluded that these cables are qualified based on the fact that they are similar to the qualified cables. The staff requests the licensee to confirm that these cables are identical to the qualified cables or provide qualification basis for these cables in accordance with 10 CFR 50.49.

NSPM Response

The cables are not identical. The basis for qualification is from Table 1 of IEEE Standard 383-1974 in which the basis of similarity of cable types can be established using representative cable configurations. The actual vendor certifications for the subject cables were identified and documented. The vendor certifications and procurement data for the specific Turbine Building cable runs all indicate environmental qualification per vendor reports currently used by Monticello per the three (3) EQ files identified in Section 3.4.5 of T1004, Revision 1. For the Rockbestos Firewall EP cables, this is Rockbestos Report QR-9201, for the Rockbestos Firewall III cables these are Rockbestos Reports QR-5804/5805, and for the Brand-Rex cables, this is Franklin Report F-C5120-1. The qualification testing/review of the identified vendor reports bounds conditions specified for the drywell or steam chase. The environmental conditions for these areas are significantly greater than those conditions predicted for the subject Turbine Building areas through which the cables are routed.

NRC RAI No. 9

In Section 3.4.5, it is not clear to the staff that the equipment in TB volumes 7 and 8 are credited for EPU conditions. The staff requests the licensee to confirm that 1) the equipment in TB volumes 7 and 8 (which are considered harsh environments due to EPU conditions) are not required to mitigate accidents under EPU conditions and 2) the failure of the equipment in TB Volumes 7 and 8 would not adversely affect the safety functions of the associated components and systems.

NSPM Response

The equipment in TB Volumes 7 and 8 are not required to be operable for the HELB events that cause the environments to become harsh, and the failure of the equipment in these volumes does not affect the components and systems that are credited to achieve safe shutdown.

NRC RAI No. 10

In Section 3.4.5, the licensee stated that 1) there is a single Valcor solenoid valve in the EQ program located in TB Volume 21 and 2) this solenoid valve is same as the qualified solenoid valve in the redundant train located in the RB; and therefore, the solenoid valve in the TB remains qualified. The staff requests the licensee to confirm that 1) the solenoid valve in the TB has been maintained in accordance with the 10 CFR 50.49 program and 2) the qualification of the Valcor valve located in the RB envelops the qualification requirements of the solenoid valve located in TB volume 21.

NSPM Response

It appears that the wording in Section 3.4.5 of Task Report T1004, Revision 1 regarding the Valcor solenoid valve is ambiguous and clarification is needed. The valve in question, Valcor solenoid valve SV-4235 located in Turbine Building Volume 21 has always been in the EQ Program. It is not being added because of EPU. The solenoid valve has always been environmentally qualified for the specified conditions at its location and is maintained as an EQ component in accordance with 10 CFR 50.49 criteria. The solenoid valve will continue to remain qualified for the conditions predicted under EPU for Turbine Building Volume 21.

NRC RAI No. 11

In Section 3.4.7, table 3.4.7-1, for GE SIS cable (or any other manufacturer type SIS cable), the licensee is crediting 50% reduction in Beta radiation due to localized shielding. The licensee further stated that there is no increase in drywell Beta radiation dose as a result of EPU. Based on this information, the staff requests the licensee to clarify why revising the analysis is necessary. Furthermore, the staff requests the licensee to provide details of localized shielding and confirm that these SIS cables are not used in open junction boxes. If these cables are used in open junction boxes, then the licensee should provide a detailed evaluation of beta radiation qualification.

NSPM Response

In lieu of a specific drywell beta dose, Monticello adopted the 2.00E+08 Rad unshielded Beta dose specified in Section 4.1, Item No. 2 of the DOR Guidelines (Enclosure No. 4 of IE Bulletin 79-01B). The beta dose reduction is in accordance with the regulatory guidelines as discussed below. This EPU approach is consistent with the current licensing basis for CLTP.

The CLTP EQ analysis for the GE SIS wire type SI-57275 credits sacrificial shielding of some portion of the insulation to achieve qualification for the 2.00E+08 Rad unshielded Beta dose. For EPU, an alternate method of beta reduction was suggested to eliminate the need for crediting a sacrificial layer for this GE SIS wire type. The alternate approach credits localized shielding and location arrangements. The only GE SIS wire type inside the drywell is that which was supplied as pigtail wiring as part of the original plant canister electrical penetrations. These penetrations are located peripherally at the drywell wall. Based on configuration, the unshielded Beta dose may be reduced by semi-infinite cloud modeling for devices located at the containment wall per Appendix D of NUREG-0588, Section 3 and Regulatory Guide 1.89, Revision 1, Appendix D, Section 3. This is the basis of adopting the alternate 50% reduction in Beta dose for the GE SIS wire at the canister electrical penetrations.

Other cables installed in the drywell (all with jackets) have credited localized shielding of 50% reduction based on potential installations in open cable trays per Item 9 of Section 1.4 of NUREG-0588. For any cable inside the drywell, the potential for exposed cable would only be at the penetrations or open cable trays. Thus, the alternate Beta dose reduction credit methods for the General Electric cables inside the drywell would be consistent with other cable types inside the drywell and not rely on sacrificial layer shielding techniques and remain in compliance with regulatory guidance for creditable Beta dose reduction factors.

NRC RAI No. 12

Based on the staff's review of Section 3.4.5, it is not clear to staff that the licensee evaluated all equipment, other than cables, in TB Volume 13. The staff requests the licensee to confirm that all equipment was evaluated for EPU conditions in TB Volume 13.

NSPM Response

Section 3.4.5 is not particularly clear on this point, and NSPM regrets the omission.

Turbine Building Volume 13 is a locked high radiation area, and an equipment walk-down was not possible prior to the issuance of Task Report T1004, Revision 1. A review of plant drawings determined that there is no equipment located in this volume within the scope of 10 CFR 50.49 as stated in the task report. A walk-down on April 21, 2009 (during the refueling outage) confirmed that no safety related electrical equipment is located within this volume.

NRC RAI No. 13

The following questions relate to the ASCO pressure switch evaluation in attachment A2:

NRC RAI No. 13(a)

In Note 2, the licensee stated that pressure switches PS-4666 and PS-4671 that are located in RB Volume 31 experience a peak HELB pressure of 17.19 psia. The licensee further stated that these switches are only required to function for inside drywell events; and therefore, they are not required to be qualified for the HELB peak pressure in RB Volume 31. The staff requests the licensee to confirm that the failure of these switches during a HELB event will have no adverse impact on the safety functions of the associated components and systems.

NSPM Response

The failure of these pressure switches during a HELB event would have no adverse impact on plant safety or mislead the plant operators.

These are the same pressure switches discussed in RAI No. 5 previously, concerning HELB temperature qualification. The "Note 2" captured on Page A2-2 was not modified for EPU. The ASCO pressure switches in Reactor Building Volume 31 only serve the Regulatory Guide 1.97 function of Containment Isolation Valve Position Indication. As indicated in the response to RAI No. 5 above, the detection and isolation equipment associated with the RWCU break and the subsequent safe shutdown are not impacted by failure of the ASCO pressure switches.

NRC RAI No. 13(b)

The licensee stated that the pressure switches located in RB Volumes 1 and 3, and "M" style switches located in RB Volume 31 have short-term function durations and are bounded by the accident test. The staff requests the licensee to confirm that the failure of these switches during the entire accident event will have no adverse impact on the safety functions of the associated components and systems.

NSPM Response

For clarification, the pressure switches located in Reactor Building Volumes 1, 3 and 14 (not 31) have short-term functions as listed in Insert Text "A" of Attachment A2 of T1004, Revision 1. Failure of these switches during the entire accident event will have no adverse impact on the safety functions of the associated components and systems.

The ASCO pressure switches located in Reactor Building Volumes 1 and 3 (PS-10-105A to D) are part of the permissive logic of ADS valves RV-2-71A/C/D (NX-7905-46-4/8 and NX-7831-143-2). Acceptance criterion number 2 of Calculation CA-94-086 and General Electric Specification 21A1060AB (MPS-0167-AB) indicate ADS function time of 10 hours. The ADS valves (RV-2-71A/C/D) can operate under automatic or manual control. For compliance with long-term depressurization of NUREG-0737, Item II.K.3.28, relief valves RV-2-71B and F function under manual control only for this need (NRC Safety Evaluation March 4, 1985, "Verify Qualification of Accumulators on ADS Valves"). The ASCO pressure switches located in Reactor Building Volumes 1 and 3 (PS-10-105A to D) are not connected to the long-term relief valves RV-2-71B or F. As such, failure of PS-10-105(A to D) beyond their 10 hour operating time has no impact on plant safety.

The ASCO pressure switches located in Reactor Building Volume 14 (PS-13-87(A to D)) sense low RCIC steam line pressure. Although listed as one of the Group 5 isolation signals per USAR Table 5.2-3b for RCIC System isolation (on low RCIC steam line pressure), the pressure switches can only detect breaches in RCIC steam line inside the drywell upstream of the venturi elbow tap (P&ID NH-36251). The RCIC System is not a credited safe-shutdown system for accident mitigation. Pressure switches PS-13-87(A to D) are included in the EQ Program for detection of RCIC System breaks inside the drywell.

NRC RAI No. 13(c)

The licensee stated the following:

"As a result of accepting twice the thermal aging for the Viton as represented by the 30 day exposure at 210°F for specimen # 3, the limiting life component has changed from the Viton to either "M" or non "M" style internal switch for all ASCO pressure switches."

The staff does not understand the intent of this discussion. The staff requests the licensee to provide a detailed explanation of the above statement.

NSPM Response

The enclosed statement on Page A2-7 of Attachment A2 was intended to indicate the qualified life change basis for EQ File 98-006, ASCO Pressure Switches and not be a part of the formal EQ File revision. The attachment was principally provided to support the interim post-accident operating time evaluation of the ASCO pressure switches under EPU conditions. It was recognized during the assessment of EPU conditions that there was a need to incorporate changes to normal plant ambient temperature conditions. The normal plant ambient temperature condition has increased for most plant areas in which ASCO pressure switches are located. The +5°F or +10°F designators within the enclosed text on Page A2-7 of Attachment A2 indicate the normal ambient temperature increases.

As part of the preliminary EQ file revision under the EPU project, it was discovered that specimen number 3 in the ASCO test (ASCO Report AQR-101083) included Viton seals that were aged for the indicated 30 days at 210°F. This represents a departure from the current EQ File of record which only credited 8.5 days of aging for the Viton material, but is a legitimate conclusion reached from further review of the ASCO testing. The current EQ file basis was conservative in choosing to use the Viton aging of 8.5 days.

The "doubling" of the aging time only applies to the ASCO pressure switches with the "M" style micro switch. In these cases, the limiting component shifted to the 15 days of aging on the "M" style micro switch versus the "newly accepted" 30 days of Viton aging. The aging for the non-M style micro switches is 30 days of aging for either the Viton or micro switch. Regardless, in all cases, the life limiting component switched from the Viton to the micro switch sub-component. The use of the wording "twice the thermal aging" or "doubling of the aging" was somewhat an unclear phrase to describe the effect of shifting the life limiting component from the Viton to the micro switch. Again, the wording within the box on Page A2-7 of Attachment 2 is not intended and will not be included as part of the formal EQ file revision. The conclusions and re-calculated lives and post-accident operability discussions otherwise presented in Attachment A2 stand correct based on the data presented therein.

Independent of the EPU project, ASCO pressure switches PS-4664 through PS-4672 were being replaced in Refueling Outage 24 as a precaution to ensure they would not exceed their qualified life based on the current EQ file analysis basis.

NRC RAI No. 13(d)

Based on our review of the licensee's ASCO pressure switch evaluation, the staff understands that the licensee has used thermal aging to qualify for the loss-of-coolant accident (LOCA) and/or Post-LOCA conditions. The staff requests the licensee to provide a detailed explanation and justification for using this methodology for qualifying these switches.

NSPM Response

The complete analysis of ASCO pressure switches required for HELB mitigation included utilization of the ASCO accident testing of 10 hours at 210°F of steam testing. To minimize the volume of the EPU Task Report, only those pages from the current EQ file being revised or amended in the interim for post-LOCA evaluation under EPU conditions were included.

In the case of EQ File CA-98-006 for ASCO pressure switches; there is an additional appendix that evaluates HELB accident conditions. The ASCO pressure switches required for HELB mitigation have operating times less than and bounded by the 10 hours of accident testing conducted by ASCO (as reviewed in Table 3.4.4-1 of Task Report T1004, Revision 1).

For long-term post-LOCA evaluation, thermal aging alone was used to envelop the qualified life plus post-LOCA operating time periods. Under a design basis LOCA, the Reactor Building is not a harsh steam environment, but is heated gradually. As a bounding location, the torus compartment, given its significant heat source (suppression pool), is the area of the Reactor Building most affected by post-LOCA heat-up conditions. Figure 3.4.6-1 of the task report shows a composite of the post-LOCA heat-up conditions of the Reactor Building. This figure is dominated by the post-LOCA conditions for the torus compartment for either EPU or CLTP plant conditions. The curve shown reveals the general slowly changing nature of post-LOCA heat-up conditions in the Reactor Building.

The interim evaluation for EPU conditions for the ASCO pressure switches as provided in Attachment A2 focused mainly on post-LOCA heat-up conditions of the torus compartment, but also provided details of other Reactor Building post-LOCA heat-up conditions as well. For EQ purposes, post-LOCA and HELB events are not concurrent. The analysis presented results in a reduction of the qualified life by the amount of expected degradation imparted to the equipment under LOCA operation. In all cases, the post-LOCA heat-up temperature is below the thermal aging temperature of 210°F. Accordingly, there is sufficient basis to qualify the ASCO pressure switches for long-term post-LOCA operation in the Reactor Building using thermal aging alone.

NRC RAI No. 14

Based on our review of the GE Cable qualification evaluation that was provided in attachment A4, the staff understands that the licensee has used thermal aging to qualify for the LOCA and/or Post-LOCA conditions. The staff requests the licensee to provide a detailed explanation and justification for using this methodology for qualifying these cables.

NSPM Response

Similar to the response to RAI No. 13(d) discussed previously, the information included in Attachment A4 of Task Report T1004, Revision 1 for the General Electric Cables was limited to that which only supported the interim Post-LOCA evaluation of the cables for EPU conditions. Within the current EQ File of record for the GE cables, there are additional analyses addressing accident qualification.

The GE cables used at Monticello are all original plant cables and qualified in accordance with the DOR Guideline criteria. The scope of GE cables used at Monticello addressed by EQ File CA-98-017, are limited to the following:

- SI-58007 and SI-58136, Butyl Rubber insulated with Neoprene jacket
- SI-57275, XLPE insulation, no jacket (SIS wire)
- SI-58109, XLPE insulation, neoprene jacket
- SI-58081, PE insulation, PVC jacket

The testing of the GE cables is limited with respect to aging and simulated accident exposure. The test conditions applicable to the GE cables are as follows:

Cable Type	Aging	Accident	Test
SI-58007 and SI-58136 Butyl Rubber	268°F for 54 days	12 hours at 300°F (or greater), followed by 12 hour decline to 200°F	Wyle Reports 44114-1/2
SI-57275 XLPE SIS	268°F for 54 days	4 days of testing, peak temperature of 340°F for 8 minutes, 330°F for next 22 minutes, followed by linear decay to 175°F over 4 day total test period.	Wyle Report 44114-2
SI-58109 XLPE insulation	268°F for 54 days	Same test data as for SI-57275 applies	Wyle Report 44114-2

Cable Type	Aging	Accident	Test
SI-58081 PE insulation	212°F for 340 hours	59 minutes at 230°F (236°F peak condition), followed by 55 hours at 217°F	Wyle Report 46830-1

The HELB/steam accident qualification of the above cables within the current EQ file of record conclude that the cables will not be exposed to transient steam temperature and pressure conditions for durations beyond those demonstrated by test in real time. As indicated in Section 5.2 of the DOR Guidelines, "accident testing [should] be at least as long as the period from initiation of the accident until the temperature and pressure service conditions return essentially to the same levels that existed before the accident." Further acceptance of shorter test conditions are permissible per the DOR Guidelines provided that the materials involved will not experience significant thermal aging during the period not tested.

The method adopted in the qualification of the General Electric cables considered all attributes of: 1) the functional time of the end device, 2) ensured that the accident testing conducted bounded the predicted transient steam conditions at the cable insulation system, and 3) reduced the permissible qualified life by the amount of time not bounded by testing. Under this scheme, the accumulated degradation anticipated under plant conditions was ensured to be bounded by the degradation imparted during the testing of the cables. In no case, are the cables installed in the plant anticipated to sustain higher levels of degradation under predicted conditions than that sustained and demonstrated by the various tests.

NRC RAI No. 15

Based on our review of attachment A17, the staff requests the licensee to provide the following information:

NRC RAI No. 15(a)

The qualification of the ITT Royal PVC cable pre-EPU conditions.

NSPM Response

The environmental qualification of the ITT Royal PVC local panel wiring is based on subjecting identical cable samples of 20 year installation to additional thermal aging and steam exposure testing. Radiation qualification is by analysis. Prior to EPU, the worst-case accident temperature condition for the ITT Royal local panel wiring was 208.1°F. The steam testing was maintained at 211°F for periods bounding the anticipated transient duration. The testing was part of the current licensing basis. It had been previously conducted to support the EQ File qualification basis of this wire type. No additional testing was performed to demonstrate qualification for EPU conditions. The testing remains acceptable for demonstrating the qualification of the wire under EPU HELB conditions as specified in the task report.

NRC RAI No. 15(b)

The results of the test on naturally aged 20 year-old ITT Royal PVC cable.

NSPM Response

As stated in the test report, the cable specimens were taken from plant locations after nearly 20 years of installation and subjected to additional thermal aging. Following the aging, all cable specimens were in good condition with no signs of visible degradation. Electrical testing included 500 Vdc insulation resistance measurements from conductor to metal mandrel. All results were greater than 2,000 megohms.

Following the thermal aging, the specimens were energized at approximately 132 Vac, 1.6 amperes while subject to condensing steam for 30-minutes. After the steam testing, the specimens were submerged in hot water for approximately 26 hours duration. The minimal insulation resistance throughout either the steam test or submergence test was 4 megohms.

NRC RAI No. 15(c)

Demonstrate the qualification of ITT Royal PVC cables for the environmental conditions (such as radiation, humidity, and pressure) where these cables are installed.

NSPM Response

The ITT Royal local panel wiring is installed in Reactor Building Volumes 14, 18, 19, 22, 33. The environmental parameters under EPU conditions for this cable type have already been evaluated in the task report under the discussion of EQ File CA-98-079. See HELB temperature per Table 3.4.4-1, post-LOCA operation under Table 3.4.6-2 (and Attachment A17), and radiation per Table 3.4.7-1 of Enclosure 17 of L-MT-08-052.