Xcel Energy®

Monticello Nuclear Generating Plant
2807 W County Rd 75
Monticello, MN 55362

May 30, 2013

L-MT-13-035
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 License No. DPR-22

Monticello Extended Power Uprate: Response to NRC Requests for Additional Information and Supplemental Information (TAC MD9990)

References:
1) Letter from T J O'Connor (NSPM) to Document Control Desk (NRC), "License Amendment Request: Extended Power Uprate (TAC MD9990)," L-MT-08-052, dated November 5, 2008. (ADAMS Accession No. ML083230111)
2) Letter from T J O'Connor (NSPM) to Document Control Desk (NRC), "License Amendment Request: Maximum Extended Load Line Limit Analysis Plus [MELLLA+]," TAC ME3145, L-MT-10-003, dated January 21, 2010. (ADAMS Accession No. ML100280558)
3) Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate: Supplement for Gap Analysis Updates (TAC MD9990)," L-MT-12-114, dated January 21, 2013. (ADAMS Accession No. ML13039A200)
4) Email from T Beltz (NRC) to J Fields (NSPM), "Monticello Nuclear Generating Plant – Draft Requests for Additional Information (SRXB) re: Review of Extended Power Uprate (MD9990)," dated March 28, 2013. (ADAMS Accession No. ML13137A103)
5) Email from T Beltz (NRC) to J Fields (NSPM), "Monticello Nuclear Generating Plant – Requests for Additional Information (SCVB) Supporting the EPU and MELLLA+ Reviews (TAC Nos. MD9990 and ME3145)," dated April 24, 2013. (ADAMS Accession No. ML13137A102)

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6) Email from T Beltz (NRC) to J Fields (NSPM), "Monticello Nuclear Generating Plant –Requests for Additional Information (EMCB) re: Extended Power Uprate License Amendment Request (TAC No. MD9990)," dated May 10, 2013. (ADAMS Accession No. ML13136A012)
7) Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate: SECY 11-0014 Use of Containment Accident Pressure – Responses to Requests for Additional Information (TAC MD9990)," L-MT-13-033, dated March 21, 2013. (ADAMS Accession No. ML13085A033)
8) Letter from T J O'Connor (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate: Updates to Docketed Information (TAC MD9990)," L-MT-10-072, dated December 21, 2010. (ADAMS Accession No. ML103570026)
9) Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate: Supplement to Revise Technical Specification Setpoint for the Automatic Depressurization System Bypass Timer (TAC MD9990)," L-MT-12-091, dated October 30, 2012. (ADAMS Accession No. ML12307A036)

Pursuant to 10 CFR 50.90, the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, 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. This is also known as an extended power uprate (EPU).

Also pursuant to 10 CFR 50.90, NSPM requested in Reference 2 an amendment to the MNGP Renewed OL and TS to allow operation within the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain.

On November 20, 2012, NSPM presented to the NRC the results of a Gap Analysis performed to verify the adequacy of the EPU documentation. Due to the delay in review of the MNGP EPU License Amendment Request (LAR), the NRC was concerned that various aspects of the NRC review were no longer applicable. Through the Gap Analysis review NSPM demonstrated that a small set of technical issues required revision and some design and licensing bases information had changed, but overall the body of EPU documentation was correct with the exception of the issues identified for correction. In Reference 3 NSPM provided to the NRC the results of many of the identified gaps and the associated corrections to EPU documentation.

In Reference 4, the Reactor Systems Branch of the NRC sent requests for additional information (RAI) concerning the letter sent by NSPM in Reference 3.

In Reference 5, the Containment and Ventilation Branch of the NRC sent a RAI concerning the letter sent in Reference 7. Enclosure 4 to this letter provides the NSPM response to the NRC RAI in Reference 5.

Enclosure 1, Part A, to this letter provides the NSPM response to the NRC RAI 1 from Reference 4. Question 6 will be addressed under a separate letter. In addition, Enclosure 1, Part B, also contains responses to RAIs 1 – 4 from Reference 6.

Enclosure 2 to this letter is General Electric-Hitachi (GEH) letter GE-MNGP-AEP-3284, Enclosure 1 which provides responses to NRC RAIs 2 - 5 from Reference 4 and RAI 10(b) from Reference 5.

Enclosure 3 provides supplemental information in the form of a revised calculation to support the Automatic Depressurization System (ADS) bypass timer TS change for the EPU LAR. NSPM described the ADS bypass timer TS change in letters L-MT-12-091 (Reference 9) and L-MT-13-019 (ADAMS Accession No. ML13037A200). NSPM discovered an error in calculation 03-036, Revision 1, and notified the NRC of the error in a telephone conference call on April 5, 2013. More details concerning the calculation change are included in Enclosure 3 including calculation 03-036, Revision 2 which corrects the error.

The RAI responses and supplemental information provided herein do not change the conclusions of the No Significant Hazards Consideration and the Environmental Consideration evaluations provided in Reference 1 as revised by References 8 and 9 for the Extended Power Uprate LAR. Further, the RAI responses and supplemental information provided herein do not change the conclusions of the No Significant Hazards Consideration and the Environmental Consideration evaluations provided in Reference 2 for the MELLLA+ LAR.

In accordance with 10 CFR 50.91(b), a copy of this application supplement, without enclosures is being provided to the designated Minnesota Official.

<u>Summary of Commitments</u>

This letter makes no new commitments and no revisions to existing commitments.

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

Executed on: May 30, 2013

Mark A. Schimmel
Site Vice-President
Monticello Nuclear Generating Plant
Northern States Power Company-Minnesota

Enclosures (3)

cc: Administrator, Region III, USNRC (w/o enclosures)
Project Manager, Monticello Nuclear Generating Plant, USNRC
Resident Inspector, Monticello Nuclear Generating Plant, USNRC (w/o enclosures)
Minnesota Department of Commerce (w/o enclosures)

ENCLOSURE 1

MONTICELLO NUCLEAR GENERATING PLANT

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

FROM THE REACTOR SYSTEMS BRANCH

AND THE MECHANICAL AND CIVIL BRANCH

This enclosure provides responses from the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy, to requests for additional information (RAI) provided by the Nuclear Regulatory Commission (NRC) Reactor Systems Branch on March 28, 2013 (Reference 1) from the Mechanical and Civil Branch dated May 10, 2013 (Reference 2).

<u>**References**</u>

1. Email from T Beltz (NRC) to J Fields (NSPM), "Monticello Nuclear Generating Plant – Draft Requests for Additional Information (SRXB) re: Review of Extended Power Uprate (MD9990)," dated March 28, 2013. (ADAMS Accession No. ML13137A103)
2. Email from T Beltz (NRC) to J Fields (NSPM), "Monticello Nuclear Generating Plant – Requests for Additional Information (EMCB) re: Extended Power Uprate License Amendment Request (TAC No. MD9990)," dated May 10, 2013. (ADAMS Accession No. ML13136A012)

<u>**PART A – Reactor Systems Branch RAI dated March 28, 2013**</u>

This section covers question 1 of the March 28, 2013 RAIs. Responses to questions 2 – 5 of the March 28, 2013 RAIs are provided in Enclosure 2. Question 6 will be addressed under a separate letter.

The NRC question is provided below in italics font and the NSPM response is provided in the normal font.

<u>***NRC Question***</u>

1. *Page 16 of Enclosure 1 to the January 21, 2013, letter (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13039A200) discusses Item 8 of the Extended Power Uprate (EPU) Gap Analysis, concerning emergency core cooling system (ECCS) pump flow rates. The response refers to additional correspondence (ADAMS Accession Nos. ML12276A057 and*

ML12276A057) which, based on cursory review, appear to indicate that some assumptions and analyses credit revised ECCS pump flow rates that remain bounded by the SAFER ECCS evaluation.

Please confirm that the SAFER ECCS evaluation includes ECCS pump flow rates that are bounding of these revised ECCS pump flow rate assumptions.

NSPM Response:
NSPM provided Emergency Core Cooling System (ECCS) pump flow rates used for the Net Positive Suction Head (NPSH) Containment Accident Pressure (CAP) evaluation of the Design Basis Accident – Loss of Coolant Accident (DBA-LOCA) in L-MT-12-082 (Reference A-1) Tables 6.6.1-1 and 6.6.1-2. These flow rates were selected based on meeting SECY 11-0014, Enclosure 1, section 6.3.6 requirement that:

"The flow rate chosen for the NPSHa analysis should be greater than or equal to the flow rate assumed in the safety analyses that demonstrate adequate core and containment cooling. This ensures that the safety analysis and the NPSH analysis are consistent."

Thus the NPSHa flow rate values selected bound the safety analysis flow rate values used in L-MT-08-052 (Reference A-2), Enclosure 5, section 2.8.5.6.2.

The SAFER ECCS evaluation ECCS pump flow rates for flow delivered to the core are unchanged. The changes shown account for changes in pump flow due to the NPSH evaluation required by SECY 11-0014. Flow rates assumed for the NPSH evaluation are greater than or equal to the flow rates required for the SAFER ECCS evaluation (ie they are bounding).

References

A-1 Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate and Maximum Extended Load Line Limit Analysis Plus License Amendment Requests: Supplement to Address SECY 11-0014 Use of Containment Accident Pressure (TAC Nos. MD9990 and ME3145)," L-MT-12-082, dated September 28, 2012. (ADAMS Accession No. ML12276A057)

A-2 Letter from T J O'Connor (NSPM) to Document Control Desk (NRC), "License Amendment Request: Extended Power Uprate (TAC MD9990)," L-MT-08-052, dated November 5, 2008. (ADAMS Accession No. ML083230111)

PART B – Mechanical and Civil Branch RAIs dated May 10, 2013

This section covers responses to questions 1 – 4 of the May 10, 2013 RAIs. The NRC question is provided below in italics font and the NSPM response is provided in the normal font.

NRC Question

1. *With regard to condensate/feedwater modification, letter L-MT-12-114, Enclosure 1, page 56/80, Item 26, indicates that a "Complete discussion regarding jet impingement and pipe whip" were requested by the NRC.*

   Page 62/80 states that piping evaluations resulted in "a new limiting (for flooding) postulated 14"-line crack at the inlet to the 14 feedwater heater." Also, "The new crack did not result in any new jet impingement or pipe whip targets." Page 62/80 shows that "Pipe whip and jet impingement analyses are pending for the Condensate pump, Feedwater pump, and piping replacement modifications."

   If the statement above regarding pipe-whip and jet-impingement is accurate (i.e., analyses are still pending), then it will need to be discussed with the licensee.

NSPM Response

The condensate pump, feedwater pump and piping replacements refer to modification work that is currently being installed in MNGP. In the statements above the use of the word "pending" was intended to clarify that the final as-built condition verification of calculation accuracy is pending until the completion of installation activities. Analyses related to design changes are approved and the new crack did not result in any new jet impingement or pipe whip targets.

To clarify our response, see Appendix A for revised pages from L-MT-12-114 (Reference B-1), Enclosure 2, Item 26 that provide clarifying information.

NRC Request

2. *If the initial information in PUSAR regarding the paragraph that discusses the 90% and 63% of the RWCU HELB M&E releases is no longer accurate and has been deleted in the revised pages of the PUSAR, then the mark-up or revision to RAI-6(a) Response needs to reflect that. As it currently exists in the submittal, it indicates that a question has been asked for the 90% and 63% increases and that an answer has not been provided.*

NSPM Response

Although the 90% and 63% increase noted did refer to EPU, those values have been superseded with subsequent mass and energy release values that incorporated enhanced characterization of actual releases. For RWCU the CLTP analysis was based on a single bounding break assumption. Whereas, the EPU analysis included consideration of multiple analyses for required break locations that included consideration of improved analysis assumptions such as double ended break flows and system depletions.

If both CLTP and EPU HELB cases were run using similar analysis assumptions, the changes in mass and energy releases would be minor as a result of EPU. The minor impact on mass and energy releases is supported by the fact that there were no significant piping changes in RWCU and process temperatures only change by 0.5°F between CLTP and the constant pressure EPU. The results of the HELB calculations for RWCU included both improved analysis assumptions and EPU conditions. The results of these calculations were tabulated by volume and provided in Tables 26-1 and 26-2 of Reference B-1.

To clarify our response, see Appendix A for revised pages from L-MT-12-114 (Reference B-1), Enclosure 2, Item 26 that provide clarifying information.

NRC Request

3. *Tables 26-1 and 26-2 contain data for HELB flood levels, HELB temperatures and HELB pressures which show increases at EPU conditions compared to CLTP. The licensee needs to evaluate these data for the increases shown and determine whether the impacted SSCs are structurally adequate to perform their intended design functions for the increases in differential pressures, temperatures and flooding levels.*

NSPM Response

Each HELB analysis evaluated flood levels to verify no acceptance criteria were exceeded. Volume (room) temperature limits were verified accordingly. Pressure differentials across walls (block walls are most limiting) were also confirmed against established wall specific limits. The Table 26-1 and 26-2 (from Reference B-1) HELB analysis parameters for building volume flooding, temperature and pressure are all acceptable, and meet corresponding structural limits.

NRC Request

4. *In the old response (2009 era) to RAI-6(b), NSPM stated that these parameters were evaluated for plant areas and structures. At that time, though, the HELB analyses and modifications were incomplete. Now that they are completed, the licensee has shown how these parameters have increased for EPU. If the effects of the increases in these parameters have been evaluated and shown to be acceptable, that is fine. The licensee needs to make that statement, since the staff does not make those determinations.*

NSPM Response
The response to RAI 6(b) in L-MT-09-044 (Reference B-2) stated that HELB analysis output parameters evaluated acceptably for plant areas and structures. Upon completion of all HELB analyses associated with the EPU, re-assessment of outputs (e.g., temperature, pressure, and water level) indicated the same conclusion; all parameters were found acceptable with regard to plant area and structural requirements.

References

B-1 Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate: Supplement for Gap Analysis Updates (TAC MD9990)," L-MT-12-114, dated January 21, 2013. (ADAMS Accession Nos. ML13039A200 and ML13039A201)

B-2 Letter from T J O'Connor (NSPM), to Document Control Desk (NRC), " Monticello Extended Power Uprate: Response to NRC Mechanical and Civil Engineering Review Branch (EMCB) Requests for Additional Information (RAIs) dated March 28, 2009 (TAC MD9990)," L-MT-09-044, dated August 21, 2009. (ADAMS Accession No. ML092390332)

Appendix A

Revised Pages from Docketed Correspondence

Included within this appendix are revised pages from marked up page changes provided in L-MT-12-114 (Reference B-1) Enclosure 2. The following pages are included:

- Pages 7, 8 and 9 (including Insert A) of 46 from L-MT-09-044, Enclosure 1
- Pages 28 and 29 of 46 from L-MT-09-044, Enclosure 1

6 pages follow

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EMCB RAI No. 6 (a)

The same paragraph on page 3-23, as above, in reference to the reactor water cleanup (RWCU), continues as follows:

> "For the break location that was analyzed during Rerate, new mass and energy release calculations considered additional blowdown sources that had not been considered in the previous 1996 analysis. This resulted in an increase in integrated mass release of about 90% and an increase in integrated energy release of 63 percent."

Confirm that the 90% and 63% increases are referring to the proposed EPU.

NSPM RESPONSE

Replace this text with the applicable portions of Table 26-2 - Reactor Building HELB Results. This table is provided in L-MT-12-114, Item 26. See also RAI response provided in L-MT-13-035, Enclosure 1, Part B - RAI 2.

~~The 90% and 63% increases are not referring to the proposed EPU. It is referring to the change in assumptions as noted in response to RAI 3 above rather than system operating condition changes resulting from EPU.~~

If the CLTP HELB cases were run using similar assumptions, the changes in mass and energy releases would be minor as a result of EPU.

As noted on PUSAR page 2-21:

> A review of the results from several recent EPU submittals concluded that, in most cases, environmental conditions are bounded by previous analyses, confirming that EPU produces relatively minor effects.

EMCB RAI RAI No. 6(b)

Please explain how the effects of the increased mass and energy release have been evaluated, include evaluations of pipe whip restraints and jet targets.

NSPM RESPONSE

Changes in mass and energy were evaluated for impacts on HELBs using the GOTHIC code. This allowed a determination of time histories for all plant areas to evaluate effects on temperature, pressure and flooding. Differential pressures between plant areas verified acceptable margins for structures such as block walls. The effects of changes to temperature, pressure and flooding have been evaluated for impact on the environmental qualification (EQ) of equipment. Upgrades to EQ files to document this evaluation ~~are in progress~~.

have been completed.

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RWCU pipe whip, jet impingement and safe shut down analyses following postulated pipe breaks or cracks are provided in USAR Appendix I. The RWCU high energy lines are located in the RWCU compartment, steam chase; MG set room, and the North West side of elevations 962' and 935' of the reactor building. There ~~are no~~ is one postulated break~~s~~ in ~~the MG set room and~~ the reactor building elevation~~s 962' and~~ 935' based on ~~seismic analysis.~~ HELB criteria. There are no pipe whip targets for the RWCU piping in the steam chase.

The safe shutdown evaluation for the RWCU compartment in Appendix I does not rely on pipe whip restraints or jet impingement shields to protect any equipment or structures. The effects of pipe whip and jet impingement in this area do not result in the loss of components required to mitigate the break and shut down the reactor. Therefore there is no impact on RWCU pipe whip and jet impingement due to EPU.

EMCB RAI No. 7

Page 2-37 states that: "The combination of stresses was evaluated to meet the requirements of the pipe break criteria. Based on these criteria, no new postulated pipe break locations were identified." For systems affected by the EPU, specifically steam (all EPU affected steam lines) and FW lines (including condensate), provide a pipe break analysis summary table (that includes the main steam increased turbine stop valve (TSV) closure transient loads in the analysis) which compares values at EPU and CLTP conditions and shows code equation stresses and CUFs compared to break limit for stresses and CUFs. Include pipe break locations and types selected for CLTP and EPU. Include lines inside and outside containment.

NSPM RESPONSE

Systems that have piping meeting the MNGP design basis criteria for classification as "High Energy" include Main Steam, Condensate, Feedwater, Residual Heat Removal (RHR), Core Spray (CS), High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Reactor Water Cleanup (RWCU), Off Gas, Control Rod Drive (CRD), Zinc Oxide Injection (GEZIP), and Standby Liquid Control (SLC). The parameters used for stress analysis in the high energy portions of these systems are unchanged due to EPU except in the Main Steam, Condensate, Feedwater, and GEZIP systems.

The Main Steam system analysis results including TSV closure loads are provided in the table below. The stress result for the Main Steam location with the maximum HELB break postulation equation result is also included in the table. The stress at that location does not meet (is less than) the current design basis criteria to require a postulated break. ~~Hence, there is no Main Steam break outside containment postulated based on stress criteria.~~ Evaluation of jet impingement from this new crack has been assessed and no safety related equipment is in the area. This new crack is bounded by other HELB cracks and breaks in the area for the impact from expected mass and energy release. Analyses related to design changes are approved and the new crack did not result in any new jet impingement or pipe whip targets. Other postulated break locations are based on configuration (e.g., terminal ends) which is not changed by EPU. Note that in the current design

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basis, specific HELB locations are not postulated inside containment. The current design basis does not include fatigue analysis of the Main Steam piping. Due to the revised analysis of the turbine stop valve closure loads, comparison to pre-EPU values is not meaningful.

The Main Steam evaluation results shown below are performed for the EPU pressure, temperature and flow parameters, including the TSV closure loads.

Main Steam Outside Containment - Maximum EPU Results (Highest Interaction Ratio):

Insert A

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

~~The maximum Feedwater system operating temperature is 397.7°F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400°F. The design pressure for this portion of the Feedwater system is unchanged by EPU. Therefore this piping is unaffected by EPU relative to HELB postulation.~~

were

were

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves ~~will be~~ re-analyzed during the Feedwater and Condensate pump and heater replacement modification process. High Energy Line Breaks and pipe whip restraints in the high energy portion of this piping ~~will be~~ evaluated at that time. GEZIP connections to the portion of the Feedwater system ~~will be~~ analyzed as part of the modification process. ~~Details of the modifications to this piping are not yet finalized.~~ The design ~~will maintain~~ stresses in the condensate and FW piping within code allowable limits of ANSI-B31.1-1977, including Winter 1978 Addenda and the requirements of USAR Chapter 12 including USAR Appendix I. ~~Confirmation that the modifications~~ are complete and meet the code allowables ~~will be provided to the NRC~~. The FW and condensate system modifications are scheduled for ~~completion during RFO25 in 2011~~.

The calculations

maintained

installation in 2013.

EMCB RAI No. 8

Enclosure 5, PUSAR Section 2.2.1.2, Liquid Line Breaks, on page 2-23 states that:

"The mass and energy releases for HELBs in the RWCU, FW, Condensate, CRD, Standby Liquid Control, and Zinc Injection (GEZIP) systems and instrument and sample

Following startup after installation of the new turbine and new FW heaters, the FW temperature increased by approximately 5°F for a portion of the FW piping, which was no longer bounded by the design temperature of 400°F for EPU operating conditions. Therefore, the affected FW piping design temperature was increased to 410°F and piping analyses were reperformed to account for the FW temperature change. All piping continues to meet code allowables.

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Insert A

Maximum Pipe Stresses (Outside Containment)

Load Combination	Service Level	Node	Stress (psi)	Allowable (psi)	Interaction Ratio
P + DW	A	TURD	7650	15000	0.51
TH Range	A	TURB	16618	22500	0.74
P + DW + TSV	B	TURC	12288	18000	0.68
P + DW + OBE*	B	X7A	14289	18000	0.79
DW+SRSS(TSV, SSE)*	D	X7A	21026	26325	0.80
HELB TH	N/A	TURB	16618	18000	0.92
HELB DW+TH+OBE	N/A	TURD	32631	30000	1.09**

*Excluding seismic category II pipe between Stop Valves and Turbine

**Indicates a HELB at this location, this load combination is used only to evaluate the need to assume a HELB and is not required to have an Interaction Ratio <1 to meet USAR requirements.

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Maximum Support Loads
MS Relief Valve Discharge Line Support RV25A-H1 (spring hanger)

Load Condition	Service Level	Node	Max Load lb	Allowable lb	IR Max/Allow	Min Load lb	Allowable lb	IR Allow/Min
DW+TH+ SRSS(TSV,SRV,OBE)	B	285	1341	1344	0.998	1162	780	0.671

Deleted per Item 11

Main Steam Outside Containment
Maximum EPU Results (Highest Interaction Ratio):

Maximum Pipe Stresses

Load Combination	Service Level	Node	Stress psi	Allowable psi	Ratio S/Allow
P+DW	B	X7A	6877	15000	0.46
TH Range	B	TURB	19441	22500	0.86
P+DW+TSV	B	268	12236	18000	0.68
DW+TSV+SRV+SSE	D	268	13795	26325	0.52
HELB DW+TH+OBE	B	TURB	27559	30000	0.92

Maximum Turbine Loads

Load Combination	Service Level	Node	Mx ft-lb	Allowable ft-lb	Ratio Mx/Allow	Mz ft-lb	Allowable ft-lb	Ratio Mz/Allow
DW	B	*	32244	413000	0.078	171446	722000	0.237
DW + TH	B	*	271321	413000	0.657	302310	722000	0.419

*Note: Loads from all turbine nodes were combined

Maximum Support Loads
Main Steam Line Support PS-16, Node 283

Load Condition	Service Level	Component	Max Load lb	Allowable lb	IR Max/Allow
DW+TH+SRSS(TSV,SRV,OBE)	B	Anchor bolt	20026	20731	0.966

Response to Part b

~~The maximum Feedwater system operating temperature is 397.7°F at EPU conditions for the Feedwater piping from the outboard containment isolation valve to the containment and inside containment. This value is bounded by the original analysis temperature of 400°F. The design pressure for this portion of the Feedwater system is unchanged by EPU. Therefore this piping is unaffected by EPU relative to HELB postulation.~~ The current design basis for Feedwater piping analysis does not include fluid transient analysis. The stress analyses for the Feedwater piping from the outboard

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Following startup after installation of the new turbine and new FW heaters, the FW temperature increased by approximately 5°F, which was no longer bounded by the design temperature of 400°F. Therefore, the FW design temperature was increased to 410°F and piping analyses were reperformed to account for the FW temperature change. All piping continues to meet code allowables.

containment isolation valve to the containment and inside containment are therefore unaffected by EPU.

have been re-analyzed for

The feedwater piping and condensate piping from the condensate pump suction to the containment isolation valves ~~will be re-analyzed during~~ the Feedwater and Condensate system modifications (reference response to RAI 7).

Installation of these modifications is currently in progress.

EMCB RAI No. 18

In accordance with Section 2.2.2 of the PUSAR, the main steam and associated piping system structural evaluation was performed to justify the operation of these systems at EPU conditions. This evaluation showed that one small bore branch line did not meet the displacement criteria. PUSAR further states that, "Additional detailed analysis will be performed to qualify this line or the piping modified prior to operation at EPU conditions."

a) Provide identification of the small bore branch line (size, system, location, function).

b) Describe the required displacement limits and their bases.

c) Since this piping analysis, with potential piping and or support modifications, is required for EPU, please discuss the reasoning for not including this information in your application. Also, indicate when necessary modifications, as needed, will be completed.

NSPM RESPONSE

a) The branch line is a 1 inch instrument sensing line located inside the primary containment. The line connects one of the differential pressure sensing ports on the D steam line flow restrictor to a containment instrument piping penetration. This line is used for flow sensing in main steam line D and serves a safety related input function to the high flow Group 1 Containment Isolation logic that will automatically isolate the MSIV's in the event of a main steam line break.

b) A differential displacement of 1/16 inch for branch connection points was used as screening criteria in the piping analysis. Those in excess of 1/16 inch were noted as outliers needing further evaluation. The basis for the 1/16 inch criteria is:

    1. The 1/16 inch displacement produces an insignificant stress in the branch line which is typically supported by a standard deadweight span (span length from run pipe nozzle connection to first support on the branch).

ENCLOSURE 2

GE-MNGP-AEP-3284, ENCLOSURE 1 – NON-PROPRIETARY

GEH RESPONSES TO RAIS SUPPORTING THE EPU AND M+ REVIEW

REACTOR SYSTEMS BRANCH QUESTIONS 2 - 5

&

CONTAINMENT AND VENTILATION BRANCH QUESTION 10(b)

10 pages follow

ENCLOSURE 1

GE-MNGP-AEP-3284

GEH Responses to RAIs Supporting the EPU and M+ Review

GEH Non-Proprietary - Class I (Public)

NRC RAI #2:

Page 20 of Enclosure 1 to the January 21, 2013, letter discusses Item 10 of the EPU Gap Analysis, concerning the effects of a final feedwater (FW) temperature change. It states, "[General Electric-Hitachi (GEH)] performed a study and determined that the impact of the FW temperature change on anticipated operational occurrences (AOOs) was negligible."

Please describe how the study was performed and provide additional information regarding the basis for this determination.

GEH Response:

GEH performed a dual reload license for Monticello Cycle 26 at 1775 MWt (CLTP) and 2004 MWt (EPU). At CLTP conditions, the limiting AOO events were evaluated using ODYN assuming feedwater temperature input of 383.0°F and 388.0°F. The impact on corrected Change in Critical Power Ratio (ΔCPR) was determined to be < 0.0045 for all events. The increase in FW temperature was shown to benefit several transients, with the difference being > -0.0048. Additionally, the limiting transient with respect to ΔCPR decreased by 0.0019, resulting in no impact to the calculated Operating Limit MCPR. All thermal and mechanical overpower results demonstrated margin to the limits.

At EPU conditions, TRACG is the licensing basis code for AOOs. Instead of explicit analysis, the TRACG AOO LTR was leveraged to quantify the sensitivity of transients to a change in FW temperature. Table 8-10, Reference 1 documents the FW temperature sensitivity on the TTNBP event with respect to DCPR/ICPR. Specifically, a change in FW temperature of -56K (-100°F) resulted in a DCPR/ICPR effect of ~0.013. The DCPR/ICPR is scaled by the ratio of FW temperature change (5/100) to obtain a more realistic value of 0.00065 DCPR/ICPR. Section 8.2.1, Reference 1 states a 0.005 DCPR/ICPR is considered 'insensitive".

The nominal FW temperature is provided on a cycle-specific basis as input to the reload licensing evaluations. Therefore, all AOOs for future reloads will be evaluated with the appropriate FW temperature.

References:

1. NEDE-32906P-A R3, Licensing Topical Report, TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses, September 2006.

NRC RAI #3:

Page 20 of Enclosure 1 to the January 21, 2013, letter discusses the effects of a final FW temperature change. It states, "GEH further concluded that sufficient margin remains in the peak dome pressure safety limit and ASME upset condition limit when accounting for this small FW temperature change."

Describe how this conclusion was reached. Explain how much margin is required to offset the effects of the final FW temperature change, how the amount of margin remaining in these limits was determined, and how MNGP will ensure that adequate margin is maintained in cycle-specific safety analyses.

GEH Response:

A qualitative evaluation was performed and the conclusion was reached using a combination of sensitivity results for non-limiting pressurization transients (e.g. turbine trip no bypass) and the margin to the dome pressure safety limit and ASME code upset condition limit. The +5°F FW temperature change increased dome and vessel bottom pressure for the non-limiting pressurization transients by ~5 psi. The FW temperature increase would impact the limiting vessel overpressure event (MSIVF) by a similar magnitude. The pressurization rate increase due to the FW temperature increase would be similar to the other non-limiting transients; however, the high-pressure RPT would occur earlier in the event offsetting some of pressure increase. The Cycle 26 EPU results for vessel overpressure demonstrated 10.9 psi margin to the dome pressure safety limit and 30.5 psi margin to the ASME code upset condition limit.

The nominal FW temperature is provided on a cycle-specific basis as input to the reload licensing evaluations. Therefore, the ASME overpressure event for future reloads will be evaluated with the appropriate FW temperature.

NRC RAI #4:

Page 20 of Enclosure 1 to the January 21, 2013, letter discusses the effects of a final FW temperature change. The applicable section describes and evaluation of the design basis accident (DBA) – loss of coolant accident (LOCA) containment response. The section does not describe the effects that the final FW temperature change could have on the DBA-LOCA ECCS evaluation.

Please explain how the EPU ECCS evaluation accounts for the final FW temperature change.

GEH Response:

Feedwater temperature changes impact the Monticello ECCS LOCA response by directly affecting the initial core coolant energy content. With higher feedwater temperature expected at EPU power (2004 MWth) and MELLLA+ flow (46.1 Mlbm/hr) conditions (e.g. 5°F above the analysis-basis value of 395.8°F to 400.8°F [Reference 1]), a corresponding increase in feedwater enthalpy yields a small increase in core coolant inlet enthalpy (less than 2.0%). A postulated large break LOCA may then cause the core to enter boiling transition at slightly earlier times along the fuel axial length, whereas a small break LOCA would see effectively no change in boiling transition behavior. Any LOCA scenario evaluated with a small increase in feedwater temperature would also experience a small increase in the cladding heatup rate early in the event due to a minor reduction in the coolant inventory heat absorption capacity. The additional energy from the higher feedwater temperature yields slightly higher cladding temperatures until ECCS provides effective cooling and inventory makeup.

The limiting ECCS LOCA scenario for Monticello is a large break in the recirculation suction line evaluated with Appendix K assumptions at Current Licensed Thermal Power (CLTP, 1775 MWth) and MELLLA flow (47.4 Mlbm/hr) conditions with LPCI injection valve failure [Reference 2, Reference 3]. The resulting Licensing Basis Peak Cladding Temperature (LBPCT) for this scenario is 2140°F.

Heat balance assessments demonstrate lower feedwater temperature values at the CLTP power level and MELLLA flow conditions when compared to EPU power and MELLLA+ or rated flow conditions. Generally, the temperature of feedwater delivered to the reactor vessel is predominately dependent upon reactor power but weakly dependent on core flow, such that the higher EPU power level amplifies the impact of the minor core enthalpy increase on the LOCA response.

An assessment performed for the Appendix K large break LOCA scenario with a 5°F increase in feedwater temperature at EPU power and rated flow conditions shows an insignificant PCT change of approximately 6°F. A similar assessment performed for the limiting case at CLTP power and MELLLA flow conditions yields a negligible difference. Therefore the LBPCT does not change and the ECCS LOCA response for Monticello is not affected by a 5°F increase in feedwater temperature at EPU power and rated, MELLLA, and MELLLA+ flow conditions. Additionally, operation at MELLLA+ conditions requires a larger setdown in the linear heat generation rate as compared to the setdown applied for operation with MELLLA conditions [Reference 3], thus assuring the limiting LOCA scenario defining LBPCT remains at CLTP power and MELLLA flow conditions.

References

1. Letter from M.A. Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate (EPU): Supplement for Gap Analysis Updates (TAC MD9990)," L-MT-12-114, dated January 21, 2013.

2. GEH Nuclear Energy, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," NEDC-33322P, Revision 3, October 2008.

3. GEH Nuclear Energy, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," NEDC-33435P, Revision 1, December 2009.

NRC RAI #5:

Page 40 of Enclosure 1 to the January 21, 2013, letter discusses Item 15 of the EPU Gap Analysis, concerning a change in the turbine bypass valve capacity value, which apparently amounted to a slight reduction, i.e., from 11.6% to 11.5%. The section states that "the evaluation of plant transients is performed on a cyclic basis for MNGP and has been completed for EPU core design using a value of 11.5% for the evaluation of transients... the results of this... evaluation are available in the MNGP cycle 26 supplemental reload licensing report..."

Please address the effects of this change with respect to the limiting ATWS overpressure events.

GEH Response:

A change in the turbine bypass capacity could only impact the Pressure Regulator Failed Open (PRFO) event. However, plants that have small TCV and turbine bypass capacities will likely not be able to depressurize the reactor down to the low pressure isolation setpoint, which would then trigger MSIV closure. Procedurally, GEH sets the demand to ~120% of rated conditions to drive plants to these pre-isolation vessel conditions. Therefore, the ATWS overpressure analysis assumes a turbine bypass capacity above the actual value. Note the turbine bypass system is not part of the ATWS overpressure mitigation. As a result, there would be no impact to the ATWS safety analysis due to the change in the turbine bypass capacity.

NRC RAI on GEH Response - RAI 10(b)

Please provide additional information in reference to NSPM letter dated March 21, 2013 (Agencywide Documents Access and Management System Accession No. ML13085A033), Enclosure 2, GEH Response - RAI 10(b).

Refer to NEDC-33322P, Revision 2, Section 2.6.3.1.1

Section 2.6.3.1.1- Short Term Gas Temperature Response

The drywell air space temperature limit is specified in Table 2.6-1. The limit is increased for EPU from 335°F to 340°F.

The GEH response to RAI 10(b) states the following:

The peak drywell temperatures reported under EPU/MELLLA conditions in Table 2.6-1 of NEDC-33322P were obtained from a long-term containment response calculation for a small steam line break accident (SBA) at 102% of EPU power and 100% core flow with the SHEX code.

NRC Staff Comment

The GEH response to RAI 10(b) appears to conflict with the NEDC-33322P Revision 2, Section 2.6.3.1.1.

The RAI response implies that the drywell gas temperature values in Table 2.6-1 of NEDC-33322P are based on a long term SBA analysis using SHEX code.

Section 2.6.3.1.1 of NEDC-33322P is referring to the short term drywell gas temperatures listed in Tables 2.6-1.

Please provide clarification as to whether the peak drywell temperatures of 335°F, 336°F, and 338°F listed in Table 2.6-1 are based on short term SBA analysis or long term SBA analysis with SHEX code.

Please provide additional clarification (e.g., footnote(s)) in Table 2.6-1 to further differentiate between long term and short term SBA analyses.

GEH Response – Question 1

The peak drywell temperatures of 335°F, 336°F, and 338°F, which are reported in Table 2.6-1 of NEDC-33322P, Revision 2 (Reference 1), were determined from long term containment analyses for a small steam line break (SBA) performed with the GEH SHEX code.

GEH Response – Question 2

A revision to Table 2.6-1 of NEDC-33322P is included for this response which expands the discussion in footnote 5 to Table 2.6-1. The expanded footnote clarifies the analysis basis for the calculated peak drywell temperatures reported in this table. Note that the footnote now identifies that the analysis basis for peak reported drywell atmosphere temperatures in this table is the same as the basis for the peak drywell wall temperatures. The expanded footnote also now reports the peak drywell temperatures obtained from the short-term DBA-LOCA recirculation suction line break analyses performed with the GEH M3CPT code.

Table 2.6-1 Containment Performance Results

Parameter	CLTP from USAR	CLTP with EPU Method[1]	EPU	Limit
Peak Drywell Pressure (psig)	39.5	43.4	44.1[2]	56[3]
Peak Drywell Temperature (°F)	335[5]	336[5]	338[2,5]	340[4]
Peak Drywell Wall Temperature (°F)	273[5]	277[5]	278[5]	281
Peak Bulk Suppression Pool Temperature (°F)	194.2	193[6]	203 / 207[7]	208[8]
Peak Wetwell Pressure (psig)	31.2	31.3	32.7	56[3]

Notes:

1. The EPU Method, which was used for the EPU analysis, uses the EPU RTP analysis method with CLTP inputs. The EPU Method includes a more bounding initial containment pressure of 3.0 psig as compared with the CLTP of the USAR, which assumed an initial containment pressure of 2.0 psig. The EPU method also assumes the initial reactor power is at 102% of the RTP.
2. Includes an increase in the assumed initial containment pressure from 2.0 psig of the method of the USAR analysis to 3.0 psig for the EPU Method.
3. The design pressure for the drywell and wetwell is 56 psig. Maximum internal pressure is 62 psig, as shown in USAR Table 5.2-1.
4. Limit for the drywell environmental temperature is increased for EPU from 335°F shown in USAR Table 5.2-8 to 340°F.
5. Peak drywell atmosphere temperatures and peak drywell wall temperatures are calculated assuming a 0.50 sqft steam break into the drywell with UCHIDA condensing heat transfer to the drywell wall to the saturation temperature at the drywell pressure, and initiation of drywell sprays at 10 minutes. The peak drywell atmosphere temperatures obtained from the short-term DBA-LOCA recirculation suction line break analysis are 285.5°F (CLTP from USAR), 290°F (CLTP with EPU Method) and 291°F (EPU).
6. Reduction in peak bulk pool temperature from 194.2°F shown in USAR Table 5.2-4- to 193°F shown above for CLTP with EPU Method is primarily due to use of a K-value that increases with increasing hot inlet water temperature.
7. The first value is the peak suppression pool temperature for the DBA LOCA with direct suppression pool cooling, 90°F service water temperature, and an RHR heat exchanger K-value that increases with increasing hot inlet water temperature. The second number is the peak suppression pool temperature for the same DBA LOCA and 90°F service water temperature, but with containment cooling using containment sprays and a constant K-value of 147 BTU/sec°F, used for NPSH evaluation.
8. The limit for peak bulk pool temperature, determined as the design temperature for the torus-attached piping, is increased for EPU from 196.7°F (Reference 19) to 208°F.

Reference:

1. GE Nuclear Energy, "Safety Analysis Report For Monticello Nuclear Generating Station Extended Power Uprate," NEDC-33322P, Revision 2, October 2008.

ENCLOSURE 3

MONTICELLO NUCLEAR GENERATING PLANT

MODIFICATION TO CALCULATION 03-036, REVISION 2

INSTRUMENT SETPOINT CALCULATION - REACTOR LOW PRESSURE PERMISSIVE BYPASS TIMER

Recently the Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy discovered an error in calculation 03-036, Revision 1, "Instrument Setpoint Calculation - Reactor Low Pressure Permissive Bypass Timer." Calculation 03-036, Revision 1 was provided to the NRC in NSPM letter L-MT-13-019, Enclosure 4 (Reference E3-1).

The identified error concerns the accuracy of the instrument loop under normal (and trip) conditions. The instrument loop accuracy under normal (and trip) conditions should have used a value of +/- 1.93 minutes. This value should have been applied in Section 6.5.1 for calculation of the Allowable Value. Instead, a value of 1.0 minutes was incorrectly used.

The Allowable Value calculated in 03-036, Revision 1 is thus incorrect and non-conservative, as calculation of the Allowable Value using the corrected 1.93 minutes random error term for loop accuracy would result in an Allowable Value (AV) <= 17.5 minutes (less than the upper AV of 18.0 minutes given in 03-036, Revision 1).

NSPM revised the calculation as follows: The Analytical Limit of <= 19.3 minutes used in 03-036, Revision 1 results in a peak clad temperature of 1500 °F. A higher Analytical Limit allows the necessary calculation revision without impacting the Allowable Value and nominal trip setpoint determined in Revision 1 and does not change the Technical Specification setpoint. Therefore, the Analytical Limit was changed to a peak clad temperature of 1700 °F. 1700 °F is well below the maximum permitted peak clad temperature of 2200 °F and is therefore acceptable. Figure 3e in NEDC-33800P (Ref. E3-1, Enclosure 2) provides the basis for the revised Analytical Limit. See the attached calculation for more details.

References

E3-1 Letter from M A Schimmel (NSPM) to Document Control Desk (NRC), "Monticello Extended Power Uprate (EPU): Response to Request for Additional Information related to Automatic Depressurization System Bypass Timer Setting (TAC MD9990)," L-MT-13-019, dated January 31, 2013. (ADAMS Accession No. ML13037A200)

31 pages follow

Xcel Energy

Calculation Signature Sheet

Document Information	
NSPM Calculation (Doc) No: 03-036	Revision: 2
Title: Instrument Setpoint Calculation - Reactor Low Pressure Permissive Bypass Timer	
Facility: ☒ MT ☐ PI	Unit: ☒ 1 ☐ 2
Safety Class: ☒ SR ☐ Aug Q ☐ Non SR	
Special Codes: ☐ Safeguards ☐ Proprietary	
Type: **Calc** Sub-Type:	

NOTE:	Print **and** sign name in signature blocks, as required.

Major Revisions	☐ N/A
EC Number: 20651	☐ Vendor Calc
Vendor Name or Code:	Vendor Doc No:
Description of Revision: Revision 2 of this calculation is being performed to correct the calculation errors identified per Attachment C of the new revision (see AR #01377658).	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☒
Prepared by: (sign) [signature] / (print) Rhon Sanderson	Date: 04-08-13
Reviewed by: (sign) Joel Beres / (print) Joel Beres	Date: 4-8-13
Type of Review: ☒ Design Verification ☐ Tech Review ☐ Suitability Review	
Method Used (For DV Only): ☒ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) Edward P. Watzl / (print) Ed Watzl	Date: 4-9-2013

Minor Revisions	☒ N/A
EC No:	☐ Vendor Calc:
Minor Rev. No:	
Description of Change:	
Pages Affected:	
The following calculation and attachments have been reviewed and deemed acceptable as a legible QA record	☐
Prepared by: (sign) / (print)	Date:
Reviewed by: (sign) / (print)	Date:
Type of Review: ☐ Design Verification ☐ Tech Review ☐ Suitability Review	
Method Used (For DV Only): ☐ Review ☐ Alternate Calc ☐ Test	
Approved by: (sign) / (print)	Date:

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy

Calculation Signature Sheet

NOTE:	This reference table is used for data entry into the PassPort Controlled Documents Module reference tables (C012 Panel). It may also be used as the reference section of the calculation. The input documents, output documents and other references should all be listed here. Add additional lines as needed by using the "TAB" key and filling in the appropriate information in each column.

Reference Documents (PassPort C012 Panel from C020)

#	Controlled* Doc? + Type		Document Name	Document Number	Doc Rev	Ref Type**	
						INPUT	OUTPUT
1	X	PROC	APPENDIX I (GE METHODOLOGY INSTRUMENTATION & CONTROLS)	ESM-03.02-APP-I	4	X	
2	X	CALC	INSTRUMENT DRIFT ANALYSIS, AGASTAT ETR14D3 TIME DELAY RELAYS	03-054	0	X	
3	X	CALC	DETERMINATION OF INSTRUMENT SERVICE CONDITIONS FOR INPUT INTO SETPOINT CALCULATION	95-027	2	X	
4	X	DRAW	CORE SPRAY SYSTEM SCHEMATIC DIAGRAM	NX-7833-21-1	78	X	
5	X	DRAW	ELEMENTARY DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM	NX-7905-46-2	81	X	
6	X	PROC	ADS SYSTEM 20 MINUTE TIMER TEST	0113-02	11		X
7	X	PROC	STOPWATCH FUNCTIONAL TEST	1318	04	X	
8	~~X~~	CALC	AUTOMATIC DEPRESSURIZATION SYSTEM BYPASS TIMER	12-046	0	X	
9	X	CALC	INSTRUMENT SETPOINT CALCULATION - AUTO BLOWDOWN INITIATION TIME DELAY RELAY	03-037	0	X	
10	X	LIC	PLANT SAFETY ANALYSIS - ACCIDENT EVALUATION METHODOLOGY	USAR-14.07	29	X	
11	X	DBD	CORE SPRAY SYSTEM	DBD-B.03.01	04		
12	X	DBD	RESIDUAL HEAT REMOVAL SYSTEM	DBD-B.03.04	06		
13	X	DRAW	S/D RESIDUAL HEAT REMOVEL SYSTEM	NX-7905-46-3	76		
14	X	DRAW	SCHEMATIC DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM	NX-7905-46-7	76		
15	X	LIC	MNGP TECHNICAL SPECIFICATIONS (AMENDMENT 171)	TECH-SPECS	168	X	
16	X	CALC	SEISMIC ANALYSIS OF AGASTAT RELAY	95-035	0	X	
17	X	CALC	AUTOMATIC DEPRESSURIZATION SYSTEM BYPASS TIMER CURRENT LICENSED THERMAL POWER	12-050	0	X	

BLY 4-1-13
EPW 4-1-13

Record Retention: Retain this form with the associated calculation for the life of the plant.

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Calculation Signature Sheet

* Controlled Doc marked with an "X" means the reference can be entered on the C012 panel in black. Unmarked lines will be yellow. If marked with an "X", also list the Doc Type, e.g., CALC, DRAW, VTM, PROC, etc.

** Mark with an "X" if the calculation provides inputs and/or outputs or both. If not, leave blank. (Corresponds to PassPort "Ref Type" codes: **Inputs / Both** = "ICALC", **Outputs** = "OCALC", **Other / Unknown** = blank)

Other PassPort Data

Associated System (PassPort C011, first three columns) **OR** **Equipment References** (PassPort C025, all five columns):

Facility	Unit	System	Equipment Type	Equipment Number
MT	1	RHR	Relay	10A-K95A
MT	1	RHR	Relay	10A-K95B
MT	1	CSP	Relay	14A-K27A
MT	1	CSP	Relay	14A-K27B

Superseded Calculations (PassPort C019):

Facility	Calc Document Number	Title

Description Codes - Optional (PassPort C018):

Code	Description (optional)	Code	Description (optional)

Notes (Nts) - Optional (PassPort X293 from C020):

Topic Notes	Text
☒ Calc Introduction	☒ Copy directly from the calculation Intro Paragraph **or** ☐ See write-up below
☐ (Specify)	

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy	Calculation Signature Sheet

Monticello Specific Information

☒ YES ☐ N/A Topic Code(s) (See MT Form 3805): RATE, NR737
☐ YES ☒ N/A Structural Code(s) (See MT Form 3805):______

Does the Calculation:

☐ YES ☒ No Require Fire Protection Review? (Using MT Form 3765, "Fire Protection Program Checklist", determine if a Fire Protection Review is required.) If YES, document the engineering review in the EC. If NO, then attach completed MT Form 3765 to the associated EC.

☐ YES ☒ No Affect piping or supports? (If Yes, Attach MT Form 3544.)

☐ YES ☒ No Affect IST Program Valve or Pump Reference Values, and/or Acceptance Criteria? (If Yes, inform IST Coordinator and provide copy of calculation.)

Record Retention: Retain this form with the associated calculation for the life of the plant.

Xcel Energy	**Design Review Checklist**

EC Number or Document Number / Title / Revision Number: 03-036, Instrument Setpoint Calculation - Reactor Low Pressure Permissive Bypass Timer, Revision 2

Verifier's Name: Joel Beres JB 4-8-03

Discipline: Engineer

DESIGN REVIEW CONSIDERATIONS:		Yes	No	N/A
1.	Were the inputs correctly selected and incorporated into design?	☒	☐	☐
2.	Are assumptions necessary to perform the design activity adequately described and reasonable? Where necessary, are the assumptions identified for subsequent re-verifications when the detailed design activities are completed?	☒	☐	☐
3.	Are the appropriate quality and quality assurance requirements specified?	☒	☐	☐
4.	Are the applicable codes, standards, and regulatory requirements including issue and addends properly identified and are their requirements for design met?	☒	☐	☐
5.	Have applicable construction and operating experience been considered?	☒	☐	☐
6.	Have the design interface requirements been satisfied?	☐	☐	☒
7.	Was an appropriate design method used?	☒	☐	☐
8.	Is the output reasonable compared to inputs?	☒	☐	☐
9.	Are the specified parts, equipment and processes suitable for the required application?	☒	☐	☐
10.	Are the specified materials compatible with each other and the design environmental conditions to which the material will be exposed?	☐	☐	☒
11.	Have adequate maintenance features and requirements been specified?	☐	☐	☒
12.	Are accessibility and other design provisions adequate for performance of needed maintenance and repair?	☐	☐	☒
13.	Has adequate accessibility been provided to perform the in-service inspection expected to be required during the plant life?	☐	☐	☒
14.	Has the design properly considered radiation exposure to the public and plant personnel?	☐	☐	☒
15.	Are the acceptance criteria incorporated in the design documents sufficient to allow verification that design requirements have been satisfactorily accomplished?	☒	☐	☐
16.	Have adequate pre-operational, subsequent periodic test and inspection requirements been appropriately specified, including acceptance criteria?	☐	☐	☒
17.	Are adequate handling, storage, cleaning, and shipping requirements specified?	☐	☐	☒
18.	Are adequate identification requirements specified?	☐	☐	☒
19.	Are requirements for record preparation, review, approval, and retention adequately specified?	☒	☐	☐
20.	Have Design and Operational Margins been considered and documented?	☒	☐	☐

COMMENTS: ☐ None ☒ Attached (Use Form QF-0528) ☐ In EC Topic Notes

Form retained in accordance with record retention schedule identified in FP-G-RM-01.

Xcel Energy

Design Review Comment Form

Sheet 1 of 1

DOCUMENT NUMBER/ TITLE: 03-036 Instrument Setpoint Calculation – Reactor Low Pressure Permissive Bypass Timer

REVISION: 2 DATE: 04-08-13

ITEM #	REVIEWER'S COMMENTS	PREPARER'S RESOLUTION	REVIEWER'S DISPOSITION
1	Add MELLLA+ to EPU operating conditions (i.e. EPU/MELLLA+) consistent with 12-046	Added MELLLA+ to references to EPU operating conditions.	All items Satisfactory JB
2	In ALT computation, sigma value should be 2	ALT computaton sigma value corrected to be "2".	
3	Add 12-050 to inputs and include statement that AL for EPU/MELLLA+ 1400 seconds from 12-046 is bounding for CLTP (12-050) in section 6.5.1	12-050 listed as Input 4.18. Section 6.5.1 discusses the fact that the EPU-MELLLA+ operating conditions are bounding.	
4	Change QF-0549 revisions to passport values (add 12-050)	Done.	

Reviewer: Joel Beer Date: 4/8/13 Preparer: [signature] Date: 04-08-13

(Rhon Sanderson)

Table of Contents – CA-03-036 – Revision 2

Document Title Number of Pages

1. Purpose

This calculation performs a setpoint calculation for the Reactor Low Pressure Permissive Bypass Timers 10A-K95A, 10A-K95B, 14A-K27A, and 14A-K27B.

Revision 0 of this calculation was performed to support the extended calibration and surveillance intervals of the time delay relays as part of the 24-Month Fuel Cycle Extension project.

Revision 1 of the calculation was performed in accordance with License Amendment 170, Input 4.14, which removed the lower allowable limit for the Reactor Steam Dome Pressure Permissive – Bypass Timer (Automatic Depressurization system(ADS) bypass timer) of *">= 18 min", previously given in Table 3.3.5.1-1 of Technical Specifications.* Removal of this lower bound from Technical Specifications allowed revision 1 to derive a time delay setpoint to support both current (CLTP) and future (EPU/ MELLLA+) operating conditions based on information provided in design inputs 4.18 and 4.13. Revision 1 changed the nominal time delay setpoint from 20 minutes to 15 minutes, for the purpose of ensuring that peak cladding temperature remains well below the 10CFR50.46 limit of 2200 deg. F for both current (CLTP) and future (EPU/MELLLA+) operating power levels.

Revision 2 of this calculation is being performed to correct calculation errors identified in revision 1. Revision 1 of this calculation had errors identified during the NRC review associated with EPU GAP Analysis review (MD9990). CAP AR # 01377658 was initiated to drive resolution of this issue (Reference 10.11). The correction of the revision 1 errors requires justification of a higher Analytical Limit for the time delay relays (20.0 minutes vs. 19.3 minutes).

2. Methodology

This calculation is performed in accordance with ESM-03.02-APP-I (Input 4.1). The General Electric Setpoint Methodology is a statistically based methodology. It recognizes that most of the uncertainties that affect instrument performance are subject to random behavior, and utilizes statistical (probability) estimates of the various uncertainties to achieve conservative, but reasonable, predictions of instrument channel uncertainties. The objective of the statistical approach to setpoint calculations is to achieve a workable compromise between the need to ensure instrument trips when appropriate, and the need to avoid spurious trips that may unnecessarily challenge safety systems or disrupt plant operation.

Drift values for the time delay relays covered by this calculation were determined in Calculation CA-03-054 (Input 4.4).

The methodology for determining instrument setpoints is not described in the USAR or its references. However, USAR Section 7.1.2.2 does state that MNGP is committed to the GE Setpoint Methodology for instrument setpoint calculations associated with safety limits and Technical Specifications.

3. Acceptance Criteria

The setpoint and instrument settings should be established such that there is a 95% probability that the constructed Analytical Limit will envelope 95% of the instrument population of interest when all applicable instrumentation uncertainties are considered.

4. Design Inputs

4.1 Engineering Standards Manual ESM-03.02-APP-I, Appendix I (GE Methodology Instrumentation & Controls), Revision 4. The ESM provides plant specific guidance on the implementation of the General Electric guidelines (Reference 10.1) and methodology (Reference 10.2).

4.2 Deleted

4.3 Monticello Component Master List (CML). The CML contains instrument information relating to the installed equipment as listed in Section 6.2.

4.4 Calculation CA-03-054, Revision 0, Instrument Drift Analysis, Agastat ETR14D3 Time Delay Relays.

$AD_{E.Random}$	±1.7% Setpoint
$AD_{E.Bias}$	+0.2% Setpoint
Calibration Interval	24 months ±25%

4.5 Deleted

4.6 Deleted

4.7 Calculation CA-95-027, Revision 2, Determination of Instrument Service Conditions for Input into Setpoint Calculations. Data obtained from this input is listed in Section 6.2. The relays included in this calculation are not listed in CA-95-027. Data for LIS-2-3-672A & C, which are also located in the Cable Spreading Room, is used for this calculation.

4.8 NX-7833-21-1, Revision 78, Core Spray System Schematic Diagram.

14A-K27A, B	Agastat ETR14D3N

4.9 NX-7905-46-2, Revision 80, Elementary Diagram Residual Heat Removal System.

10A-K95A, B	Agastat ETR14D3N

4.10 Tyco Electronics, Agastat Nuclear Qualified Control Relays - Series EGP/EML/ETR, 4/24/2002 Edition (Attachment A).

ETR14D3N relay	125 VDC, 1 to 30 minutes
Repeat Accuracy - Normal Conditions	±5% Setpoint
Repeat Accuracy - Adverse Conditions	±10% Setpoint

The environments for which the relays are expected to trip are similar to the vendor defined normal operating conditions of the relay. However, the instrument operating range minimum temperature is 60 degrees F (Input 4.7) versus the vendor-specified normal environment minimum temperature of 70 degrees F. Therefore, for conservatism, the accuracy for adverse conditions (+/-10% of setpoint) will be applied in this calculation.

4.11 0113-02, Revision 11, ADS System 20 Minute Timer Test.

As Found Range	< 21.7 minutes
As Left Range	≥ 19 and ≤ 21 minutes

4.12 1318, Revision 4, Stopwatch Functional Test.

Maximum Allowed Deviation in Test	0.1% Reading (0.06 Sec in 1 Min Test)

4.13 Calculation 12-046, Rev 0, MNGP Automatic Depressurization System Bypass Timer Extended Power Uprate. Data obtained from this input was used to determine an acceptable nominal setpoint to ensure peak cladding temperature

(PCT) was limited to well below the 10CFR50.46 PCT of 2200°F (approximately 1700°F) for current and future EPU operating conditions. Figure 3-e of calculation 12-046 shows fuel clad temperature vs. time for the limiting RWCU break analysis.

4.14 License Amendment 170 – Removal of lower allowable limit for 'Reactor Steam Dome Pressure Permissive – Bypass Timer (Pump Permissive)'. (Attachment B is the coversheet).

4.15 Calculation 03-037 Rev 0, Instrument Setpoint Calculation ADS Blowdown Initiation Time Delay Relay

4.16 USAR-14.07 Rev 29, Table 14.7-12 - ECCS Injection Timing Parameters Used in ECCS Performance Evaluations

4.17 Calculation 95-035 Rev 0, Seismic Analysis of Agastat Relay

4.18 Calculation 12-050, Rev. 0, MNGP Automatic Depressurization System Byypass Timer Current Licensed Thermal Power (CLTP). Data obtained from Figure 3-e of 12-050 shows that the limiting RWCU break analysis for EPU has a fuel clad temperature time response that bounds the time response for CLTP operating conditions (see Input 4.13).

5. Assumptions

None.

6. Analysis

6.1 Instrument Channel Arrangement

Channel Diagram:

Definition of Channel: Each time delay relay is initiated by the one-of-two-twice low-low reactor water level signal. After the time delay, the relay provides a contact closure to the Core Spray and RHR systems (Input 4.8; References 10.5 and 10.6).

6.2 Instrument Definition and Determination of Device Error Terms

6.2.1 Device 1

6.2.1.1 Instrument Definition

		Reference
Component ID	10A-K95A, B and 14A-K27A, B	
Location:	Admin Building, 939', CSR Panels C-32 and C-33	4.3
Manufacturer:	Agastat	4.8, 4.9
Model Number:	ETR14D3N	4.8, 4.9
Upper Range limit:	30 minutes	4.10
Adjustable Range:	1-30 minutes	4.10
Input Signal:	Contact Closure	4.8, 10.5, 10.6
Output Signal:	Contact Closure	4.8, 10.5, 10.6

6.2.1.2 Process and Physical Interfaces

Calibration Conditions:		**Reference**
Temperature:	65 to 90°F	4.7
Surveillance Interval:	30 months	4.4

Calibration of the time delay relays is required every operating cycle per Input 4.4. A surveillance interval of 30 months (24 months + 25%) is used in accordance with the guidance in Generic Letter 91-04 (Reference 10.8).

Normal Plant Conditions:		**Reference**
Temperature:	60 to 104°F	4.7
Radiation:	Negligible	4.7
Pressure:	Ambient	4.7
Humidity:	0 to 90%	4.7

Trip Environment Conditions:		**Reference**
Temperature:	104°F	4.7
Radiation:	Negligible	4.7
Pressure:	Ambient	4.7

Humidity:	100%	4.7

Seismic Conditions:		**Reference**
OBE Prior to Function	1.476 g	4.17
OBE During Function	1.476 g	4.17

Process Conditions:		**Reference**
During Calibration:	N/A	N/A
Worst Case:	N/A	N/A
During Function:	N/A	N/A

These relays are not subjected to process conditions (static pressure, overpressure, elevated temperatures, etc.) that would affect the accuracy of the instrument.

6.2.1.3 Individual Device Accuracy

Term	**Value**	**Sigma**	**Reference**
VA:	$\pm$ 10.0% Setpoint (adverse) $\pm$ 5.0 % Setpoint (normal)	2 2	4.10
ATE:	0		Note 1
OPE:	N/A		Note 2
SPE:	N/A		Note 5
SE:	0		Note 4
RE:	0		Note 7
HE:	0		Note 6
PSE:	N/A		Note 3
REE:	N/A		Note 3

Note 1: Accuracy Temperature Effect (ATE) data is not specified for these relays. The ATE is considered part of the Vendor Accuracy since the operating conditions are enveloped by the vendor's qualification limits for operation in adverse conditions.

Note 2: Overpressure Effects (OPE) are not applicable to relays.

Note 3: Error effects due to Power Supply Effects (PSE) and RFI/EMI Effects (REE) are considered negligible for bi-stable electro-mechanical devices (Reference 10.1).

Note 4: Seismic Effects (SE), Section 6.2.1.2 notes the seismic conditions for the relay. These conditions are bounded by the seismic qualified provided by the vendor as described by Input 4.10. Therefore, inaccuracies due to seismic effects are considered to be included in the VA for trip conditions.

Note 5: Static Pressure Effects (SPE) do not apply to bi-stable electro-mechanical devices (Reference 10.1).

Note 6: The normal operating conditions of the relays are within the vendor specified operating range of the relay (Input 4.10). Although Input 4.7 gives a humidity of 100% for trip conditions, this is based on assumption and for the applicable Cable Spreading Room environment humidity levels would not be expected to result in condensation. There are no significant instrument accuracy effects that would result from higher (non-condensing) levels of relative humidity. Therefore Humidity Effects (HE) are considered to be included in the VA for trip conditions.

Note 7: Radiation Effects (RE) is not specified for these relays, they are considered to be included in the VA for trip conditions

$$\text{VA} = \text{Vendor Specifications (Adverse Conditions)} = 10\% \text{ of setpoint per Input } 4.10$$

$$\text{VA} = \pm 0.10 \times 20.0 \text{ minutes} = \pm 2.00 \text{ minutes} \text{ ; Note that the setpoint is conservatively assumed to be the Analytical Limit of 20.0 minutes}$$

$$A_{LN} = 2 \times \sqrt{\left(\frac{VA}{n}\right)^2 + \left(\frac{ATE}{n}\right)^2 + \left(\frac{OPE}{n}\right)^2 + \left(\frac{SPE}{n}\right)^2 + \left(\frac{SE}{n}\right)^2 + \left(\frac{RE}{n}\right)^2 + \left(\frac{HE}{n}\right)^2 + \left(\frac{PSE}{n}\right)^2 + \left(\frac{REE}{n}\right)^2}$$

$$A_{LN} = 2 \times \sqrt{\left(\frac{2.00}{2}\right)^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2 + 0^2}$$

A_{LN} = +/- 2.00 minutes

A_{LT} = +/- 2.00 minutes, as the vendor-specified adverse / abnormal environmental conditions bound the operating environment (with the exception of humidity, see discussion in Note 6 of this Section.

6.2.1.4 Individual Device Drift

Term	Value
VD:	Not Specified
DTE:	Not Specified

Vendor drift (VD) is not specified. A Monticello specific drift analysis of Agastat ETR14D3 time delay relays was performed (Input 4.4) to determine the 30 month Analyzed Drift Value (AD) for these transmitters. The AD is used in place of both the VD and the DTE (Drift Temperature Effect):

$$AD_{E.Random} = \pm 1.7\% \text{ Setpoint}$$
$$AD_{E.Bias} = +\,0.2\% \text{ Setpoint}$$

$$D_{L.Random} = AD_{E.Random} \text{ x Setpoint}$$
$$D_{L.Random} = \pm\, 0.017 \times 20.0 \text{ minutes}$$
$$D_{L.Random} = \pm\, 0.34 \text{ minutes}$$

$$D_{L.Bias} = AD_{E.Bias} \text{ x Setpoint}$$
$$D_{L.Bias} = +\,0.002 \times 20.0 \text{ minutes}$$
$$D_{L.Bias} = +\,0.04 \text{ minutes}$$

6.2.1.5 As-Left Tolerance (ALT)

Per Input 4.1, a suggested ALT is determined with the following equation:

$$ALT = \pm 2\sqrt{\left(\frac{VA_i}{n}\right)^2 + \left(\frac{C_i}{n}\right)^2 + \left(\frac{C_{iSTD}}{n}\right)^2}$$
$$ALT = \pm 2\sqrt{\left(\frac{1.00}{2}\right)^2 + \left(\frac{0.02}{3}\right)^2 + \left(\frac{0.02}{3}\right)^2}$$
$$ALT = \pm 1.00 \text{ minutes}$$

Note that vendor accuracy (VA) used to calculate the As Left Tolerance is the typical 5% of setpoint accuracy specified by the vendor for normal operating conditions. The setpoint is assumed to be the AL of 20.0 minutes.
The existing As-Left Tolerance specified in the surveillance procedure 0113-02 (Input 4.11) is ± 1.0 minute; the existing ALT of 1.0 minute will remain unchanged. A value of 1.0 minute is reasonable considering the expected 5% of setpoint accuracy expected at typical calibration conditions.

As Left Tolerance (ALT) = +/- 1.0 minutes

6.2.1.6 Device Calibration Error

Term	Value	Sigma	Reference
C_1:	0.02 minutes	3	Note 1
C_{1STD}:	0.02 minutes	3	Note 2
ALT:	1 minute	2	Section 6.2.1.5

Note 1: The Calibration Tool Error (C_1) is considered equal to the As Found tolerance from the functional test procedure (Input 4.12):

$C_1 = \pm 0.1\% x\text{Reading}$
$C_1 = \pm 0.001 x 20.0$ minutes
$C_1 = \pm 0.02$ minutes

Note 2: In accordance with Input 4.1, the calibration standard error (C_{1STD}) is considered to be equal to C_1.

Since calibration term values are controlled by 100% testing, they are assumed to represent 3-sigma values. Individual calibration error terms are combined using the SRSS method and normalized to a 2-sigma confidence level.

$$C_L = \pm 2 \times \sqrt{\sum \frac{C_1}{n}^2 + \sum \frac{C_{1STD}}{n}^2 + \frac{ALT}{n}^2}$$

$$C_L = \pm 2 \times \sqrt{\frac{0.02}{3}^2 + \frac{0.02}{3}^2 + \frac{1}{2}^2}$$

$$C_L = \pm 1 \text{ minute}$$

6.3 Determination of Primary Element Accuracy (PEA) and Process Measurement Accuracy (PMA)

There are no PEA or PMA inaccuracies associated with these relays.

PMA = 0
PEA = 0

6.4 Determination of Other Error Terms

Term	Value
Indicator Readability/Operator Reading Error (ORE)	0.02 minutes
Resistors, Multiplexers, etc.	0
Software Errors	0
Degradation of Insulation Resistance (IRE)	0

An ORE equal to the Calibration Tool Error is applied for readability and operator reaction time.

6.5 Calculation of Allowable Value and Operating Setpoint

6.5.1 Allowable Value (AV):

From Input 4.13, it can be seen that for the bounding scenario of a RWCU break at EPU/MELLLA+ operating conditions with a gate valve a time of 1579 seconds is required to reach 2200° F. Input 4.13 also shows that it takes approximately 1400 seconds to reach 1700° F. Input 4.18 shows that the EPU / MELLLA+ case bounds current operating power (CLTP) conditions with respect to clad temperature vs. time for the limiting RWCU break. In order to provide sufficient time to cool the core, the actuation of ADS should occur prior to reaching 2200° F, therefore additional consideration is given for the following delays in the ADS initiation: from Input 4.13 a delay of 36 seconds from time 0 of the scenario is taken for initiation of the low low level signal, from Input 4.16 the time required for ECCS pumps to reach rated speed is 18 seconds, from Input 4.15 the ADS timer delay is 138 seconds. Therefore an analytical value of 1208 seconds will be used.

1400s – 36s – 18s – 138s = 1208s

An Analytical Limit of 1208 seconds ensures actuation of ADS at approximately 1700 degrees F, well before reaching the 2200 degrees F limit. The Analytical Limit will be defined as 1200 seconds, or 20.0 minutes.

Analytical Limit (AL): < 20.0 minutes

Term	Value (Minutes)	Sigma	Reference
A_{LT}	2.0	2	Section 6.2.1.3
C_L	1.0	2	Section 6.2.1.6
PMA	0	2	Section 6.3
PEA	0	2	Section 6.3
IRE	0	N/A	Section 6.4
ORE	0.02	2	Section 6.4

$$AV = AL - (\frac{1.645}{2})(\sqrt{A_{LT}{}^2 + C_L{}^2 + PMA^2 + PEA^2 + IRE^2 + ORE^2}) + \text{bias terms}$$

$$AV = 20.0 - (\frac{1.645}{2})(\sqrt{2.0^2 + 1.0^2 + 0^2 + 0^2 + 0^2 + 0.02^2}) + 0$$

$$AV = 20.0 - 1.84$$

$$AV = 18.16 \text{ minutes}$$

As a result of CR 02001013 (Reference 10.9), a new Technical Specification Trip Setting is chosen to bound the As Found values (Refer to Section 6.5.5.). Conservatively rounding down the calculated AV, the Technical Specification AV per this calculation will be:

$$AV <= 18.0 \text{ minutes}$$

6.5.2 Nominal Trip Setpoint (NTSP)

Term	Valve (Minutes)	Sigma	Reference
A_{LT}	$\pm$ 2.0	2	Section 6.2.1.3
$D_{L.Random}$	$\pm$ 0.34	2	Section 6.2.1.4
$D_{L.Bias}$	$\pm$ 0.04	NA	Section 6.2.1.4
C_L	$\pm$ 1.0	2	Section 6.2.1.6
PMA	0	2	Section 6.3
PEA	0	2	Section 6.3
IRE	0	NA	Section 6.4
ORE	$\pm$ 0.02	2	Section 6.4
RAV_{Bias}	0	NA	Section 6.5.1

$$NTSP_1 = AL - \left(\frac{1.645}{2}\right)\left(\sqrt{A_{LT}^2 + C_L^2 + D_{L.Random}^2 + PMA^2 + PEA^2 + ORE^2 + IRE^2}\right) - RAV_{Bias} - D_{L.Bias}$$

$$NTSP_1 = 20.0 - \left(\frac{1.645}{2}\right)\left(\sqrt{2.0^2 + 1.0^2 + 0.34^2 + 0^2 + 0^2 + 0.02^2 + 0^2}\right) - 0 - 0.04$$

$$NTSP_1 = 20.0 - 1.91$$

$$NTSP_1 = 18.09 \text{ minutes}$$

6.5.3 Licensee Event Report (LER) Avoidance Evaluation

The purpose of the LER Avoidance Evaluation is to assure that there is sufficient margin provided between the AV and the NTSP to reasonably avoid violations of the AV. Any Z value greater than 1.29 provides sufficient margin between the NTSP and the AV. Therefore, $NTSP_2$ is calculated to provide an upper bound for the NTSP based on LER avoidance criteria.

$$Sigma^+(LER) = +\left(\frac{1}{2}\right)\left(\sqrt{A_{LN}^2 + C_L^2 + D_{L.Random}^2}\right) + D_{L.Bias}$$

$$Sigma^+(LER) = +\left(\frac{1}{2}\right)\left(\sqrt{2.0^2 + 1.0^2 + 0.34^2}\right) + 0.04$$

$$Sigma^+(LER) = +1.171$$

$$NTSP_2 = AV - (Z \times Sigma^+(LER))$$

$$NTSP_2 = 18.0 - (1.29 \times 1.171)$$

$$NTSP_2 = 16.48$$

Therefore, an $NTSP_2 \leq 16.48$, will result in a Z greater than 1.29 and provide sufficient margin between the NTSP and the Allowable Value.

6.5.4 Selection of Operating Setpoint

$$TS = NTSP_2 - ALT$$

$$TS = 16.48 - 1.0$$

$$TS = 15.48$$

The setpoint will be rounded down to 15 minutes for added conservatism.

NTSP = 15.0 minutes

6.5.5 Establishing As-Found Tolerance (AFT)

An As-Found Tolerance is calculated to provide suggested limits for use during the surveillance testing:

$$AFT = \pm\frac{3}{2}\sqrt{VA^2 + D_{L.Random}{}^2 + DTE^2} + D_{L.Bias}$$

$$AFT = \pm\frac{3}{2}\sqrt{1.0^2 + 0.34^2 + 0^2} + 0.04$$

$$AFT = -1.54 \text{ minutes}, +1.62 \text{ minutes}$$

The As Found Tolerance (AFT) range will be specified as:

AFT = +/- 1.5 minutes

Note that vendor accuracy (VA) used to calculate the As Found Tolerance is the typical 5% of setpoint accuracy specified by the vendor for normal operating conditions. The setpoint is assumed to be the AL of 20.0 minutes.

A review of As-Found data (Input 4.4) shows that these relays have historically performed within the calculated AFT.

6.5.6 Required Limits Evaluation

The purpose of a Required Limits Evaluation is to assure that the combination of errors present during calibration of each device in the channel is accounted for while allowing for the possibility that the devices may not be recalibrated. Since Leave Alone Zones are not used at MNGP, the devices are always verified or recalibrated to be within the As Left Zone. Therefore, a Required Limits Evaluation as discussed in the GE methodology is not applicable. Because the calibrated portion of this instrument loop consists only of the timers, the Loop As Found Tolerance is equal to the AFT from Section 6.5.5 above.

Loop AFT = AFT = +/- 1.5 minutes

As a result of Condition Report (CR) 02001013 (Reference 10.9), the As Found values are reviewed to verify that the As Found value is not outside the Technical Specification range. The As Found limits are not outside the Technical Specification range, and are therefore acceptable as determined in Section 6.5.5.

6.5.7 Spurious Trip Avoidance Evaluation

A spurious trip avoidance evaluation is performed to assure that there is a reasonable probability that spurious trips will not occur using the selected setpoint. The margin of the 15.0 minute setpoint to the minimum time delay of 10 minutes is large with respect to instrument accuracies. Spurious trip margin is more than adequate; no formal analysis is required.

6.5.8 Elevation Correction

None.

6.5.9 Determination of Action Setpoint

The nominal setpoint of 15.0 minutes will be used.

7. Conclusions

The results of the calculations are as follows:

Term	Value (minutes)	Section
A_{LN}:	$\pm$ 2.0	6.2.1.3
A_{LT}:	$\pm$ 2.0	6.2.1.3
$D_{L.Random}$:	$\pm$ 0.34	6.2.1.4
$D_{L.Bias}$:	$\pm$ 0.04	6.2.1.4
ALT:	$\pm$ 1.0	6.2.1.5
C_L:	$\pm$ 1.0	6.2.1.6
PEA:	NA	6.3
PMA:	NA	6.3
AV (calculated):	<= 18.0	6.5.1
$NTSP_2$:	16.48	6.5.3
Current Trip Setting:	20 $\pm$ 1.0	4.13
Proposed Trip Setting:	15 $\pm$ 1.0	6.5.1
AFT:	+/- 1.5	6.5.5
AF Limits:	>=13.5 , < =16.5	6.5.5
Elevation Correction:	NA	6.5.8

8. Future Needs

1. Revise procedure 0113-02 and supporting documentation as listed on the ADL of EC 20651 to reflect new setpoint. Revise EC 20651 as necessary due to revision 2 of this calculation. The instruments' nominal setpoints, setting tolerances, and the Allowable Value remain unchanged versus revision 1 of this calculation.

9. Attachments

Attachment A: Agastat Datasheet for EGP/EML/ETR Series Relays
Attachment B: Excerpt from License Amendment 170

10. References

10.1 GE-NE-901-021-0492, DRF A00-01932-1, Setpoint Calculation Guidelines for the Monticello Nuclear Generating Plant, October 1992.

10.2 General Electric Instrument Setpoint Methodology, NEDC-31336P-A, September 1996.

10.3 DBD B.03.01, Revision 4, Design Bases Document for Core Spray System.

10.4 DBD B.03.04, Revision 6, Design Bases Document for Reactor Heat Removal.

10.5 NX-7905-46-3, Revision 76, Schematic Diagram Residual Heat Removal System.

10.6 NX-7905-46-7, Revision 76, Schematic Diagram Residual Heat Removal System.

10.7 NEDC-32514P, Revision 1, October 1997, Monticello SAFER/GESTER-LOCA Loss-of-Coolant Accident Analysis.

10.8 Generic Letter 91-04, Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.

10.9 Condition Report 02001013, Documentation of NRC Resident Question Regarding the Application of Tech Spec Deviations in As-Found Acceptance Criteria.

10.10 Amendment No. 62 to DPR-22, Dated 03/31/89, Reactor Vessel Level Instrumentation, ADS Logic Changes and S/RV Discharge Pipe Pressure Switch Setpoints.

10.11 CAP AR # 01377658, Errors in Calc 03-036 ADS Bypass Timer Setpoint, 04-05-13

CA-03-036 REV X 2
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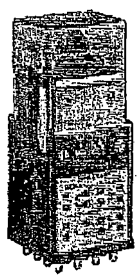
AGASTAT

4/24/2002 Edition

Nuclear Qualified Control Relays – Series EGP/EML/ETR



SEISMIC AND RADIATION TESTED

In order to satisfy the growing need for electrical control components suitable for class 1E service in nuclear power generating stations, AGASTAT® control relays have been tested for these applications. Series EGP, EML and ETR have demonstrated compliance with the requirements of IEEE Standards 323-1974 (Standard for qualifying Class 1E Equipment for Nuclear Power Generating Stations) and IEEE Standard 344-1975 (Seismic Qualification for Nuclear Power Generating Stations). Testing was also referenced to ANSI/IEEE C37.98 (formerly IEEE Standard 501-1978, Standard for Seismic Testing of Relays).

The design of Series EGP, EML and ETR control relays has evolved over 20 years of continual use in a wide range of industrial applications. Power Relay, Magnetic Latch and Timing Relay versions are available for use with a choice of coil voltages, as well as an internal fixed or adjustable potentiometer in the Series ETR time delay version.

TEST PROCEDURE

Test Procedure
AGASTAT® control relay Series EGP, EML and ETR were tested in accordance with the requirements of IEEE STD. 323-1974 (Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations), IEEE STD. 344-1975 (Seismic Qualification for Nuclear Power Generating Stations) and referenced to ANSI/IEEE C37.98 (formerly IEEE Standard 501-1978, Standard for Seismic Testing of Relays). The relays were tested according to parameters which, in practice, should encompass the majority of applications. Documented data applies to relays which were mounted on rigid test fixtures. The following descriptions of the tests performed are presented in their actual sequence.

Radiation Aging
Relays were subjected to a radiation dosage of 2.0×10^5 Rads, which is considered to exceed adverse plant operating requirements for such areas as auxiliary and control buildings.

Cycling with Load Aging
The radiated units were then subjected to 27,500 operations at accelerated rate, with one set of contacts loaded to 120VAC, 60Hz at 10 amps; or 125VDC at 1 amp, and the number of mechanical operations exceeding those experienced in actual service.

Temperature Aging
This test subjected the relays to a temperature of 100°C for 42 days, with performance measured before and after thermal stress.

Seismic Aging
Sufficient interactions were performed at levels less than the fragility levels of the devices in order to satisfy the seismic aging requirements of IEEE STD 323-1974 and IEEE STD 344-1975.

Seismic Qualification
Artificially aged relays were subjected to simulated seismic vibration, which verified the ability of the individual device to perform its required function before, during and/or following design basis earthquakes. Relays were tested in the non-operating, operating and transitional modes.

Hostile Environment
Since the relays are intended for use in auxiliary and control buildings, and not in the reactor containment areas, a hostile environment test was performed in place of the Loss of Coolant Accident (LOCA) test. Relays were subjected to combination extreme temperature/humidity plus under/over voltage testing to prove their ability to function under adverse conditions even after having undergone all the previous aging simulation and seismic testing. The devices were operated at minimum and maximum voltage extremes: 85 and 120 percent of rated voltage for AC units, and 80 and 120 percent of rated voltage for DC units, with temperatures ranging from 40°F to 172°F at 95 percent relative humidity.

Baseline Performance
In addition to aging tests, a series of baseline tests were conducted before, and immediately after each aging sequence, in the following areas:
- Pull-in Voltage
- Drop-out Voltage
- Dielectric Strength at 1650V 60Hz
- Insulation Resistance
- Operate Time (milliseconds)
- Recycle Time (milliseconds)
- Time Delay (seconds) } Series ETR only
- Repeatability (percent) } Series ETR only
- Contact Bounce (milliseconds at 28VDC, 1 amp.)
- Contact Resistance (milliohms at 28VDC, 1 amp.)

Data was measured and recorded and used for comparison throughout the qualification test program in order to detect any degradation of performance.

The SRS shape (at 5 percent damping), is defined by four points:
point A = 1.0 Hz and an acceleration equal to 25 percent of the Zero Period Acceleration (ZPA)
point D = 4.0 Hz and 250 percent of the ZPA
point E = 16.0 Hz and 250 percent of the ZPA
point G = 33.0 Hz and a level equal to the ZPA

SPECIMEN 13, 15 & 16 (EGP SERIES)
RELAY STATE: NON-OPERATE MODE (DE-ENER.)
TEST RUN NO. 318, 319, (205-206), (198-199)
AXIS (H + V):
COMPOSITE OF FB/V-, SS/V, FB/V+ X .707
DUE TO 45° INCLINATION OF TEST MACHINE.

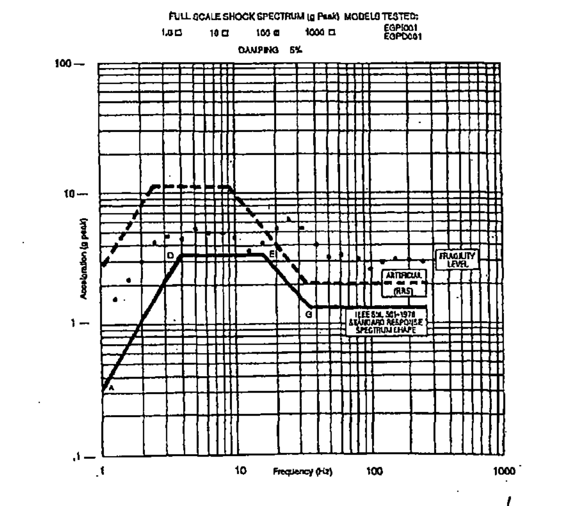


Figure 1. Model EGP, Response Sectrum, Non-Operate Mode

Additional Seismic Response Curves are available on request.

Relay State: Non-Operate Mode (De-ener.)
Test Run No. 318, 319, (205-206), (198-199)

tyco / Electronics

Specifications subject to change
Dimensions are for reference only.

Technical Support Center 1-800-522-6752
www.tycoelectronics.com

CA-Φ3-Φ36 REV ~~1~~ 2
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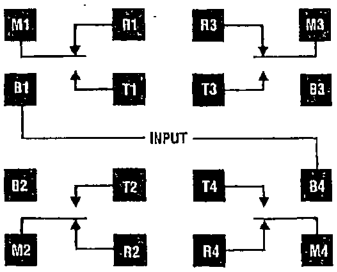
AGASTAT

Nuclear Qualified Control Relays

OPERATION

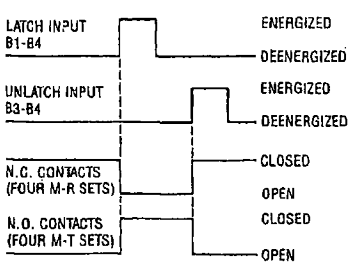
Series EGP
Power Relay

Applying a continuous voltage to the coil (B1-B4) energizes the coil and instantaneously transfers the switch, breaking the normally closed contacts (M1-R1, M2-R2, M3-R3, M4-R4) and making the normally open contacts (M1-T1, M2-T2, M3-T3, M4-T4). The contacts remain in this transferred position until the coil is deenergized, at which time the switch instantaneously returns the contacts to their original position.



Series EML
Magnetic Latch

Application of a voltage to the latching input (B1-B4) will cause the relay to latch in (Make the N.O. Contacts, break the N.C. Contacts). When this voltage is removed, the relay will remain in this "Latched" condition. Application of a voltage to the un-latching input (B3-B4) will cause the relay to dropout (Break the N.O. Contacts, make the N.C. Contacts). When this voltage is removed, the relay will remain in this "Unlatched" condition.

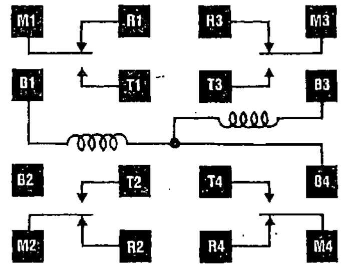


Wiring Diagram (Wiring and Connections)

The ML relay has three terminals for the windings: latching winding between terminals B1 and B4, un-latching winding between terminals B3 and B4.

The ML Relay is not symmetrical due to its three coil connections

The relays are normally delivered polarized so that terminal B4 carries the negative voltage. To reverse the polarity, a deenergize/energize cycle should be carried out using a voltage 50% greater than the normal rating.



Continuous Duty Wiring

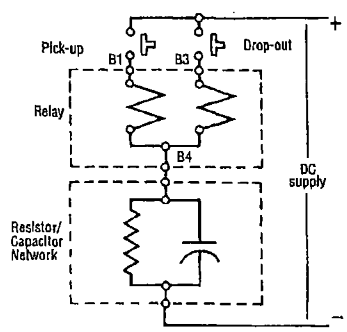
Since the double wound coil does not have a continuous duty rating, voltage pulses to the coils should not exceed a ratio of 40% on, to 60% off, with maximum power-on periods not to exceed 10 minutes.

If continuous energizing only is available, a resistor/capacitor network should be connected as shown below. In this case the shortest time between two operations must not be less than 5 seconds.

The relay will always assume the energized position in the event of both windings being energized simultaneously.

It is advisable not to put another load in parallel with the windings of the ML relay.

ML Series Relay for DC operation with a resistor/capacitor network

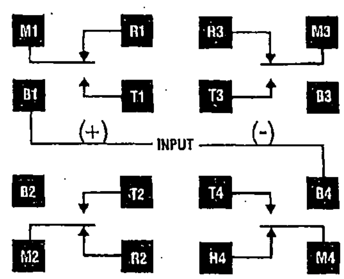


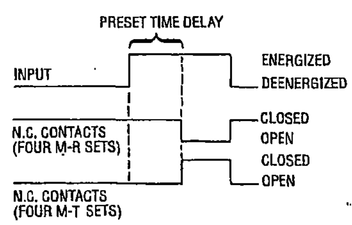
R-C Values

Nominal Voltage VDC	R OHMS ±5%	R Watts	C UF	C VDC
12	62	2	5000	15
24	240	2	2000	50
48	1000	2	500	100
125	6200	2	150	150

Series ETR
Time Delay Relay
(Delay on Energization)

Applying a continuous voltage to the input terminals (B1-B4) starts a time delay lasting for the preset time period. During this period the normally closed contacts (Four M-R sets) remain closed. At the end of the delay period, the normally closed contacts break and the normally open contacts (Four M-T sets) make. The contacts remain in this position until the relay is deenergized, at which time the contacts instantaneously return to their normal position. Deenergizing the relay, either during or after the delay period will recycle the unit within .075 second. It will then provide a full delay period upon reenergization, regardless of how often the voltage is interrupted before the unit has been permitted to "time-out" to its full delay setting.





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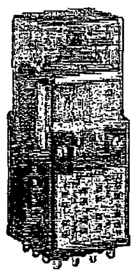
Nuclear Qualified Control Relays



SPECIFICATIONS

Contact Ratings – Series EGP/EML/ETR

Contact Capacity in Amperes (Resistive)

Contact Voltage	Min. 1,000,000 Operations
24 vdc	10.0 amps
125 vdc	1.0 amp
120 vac, 60 Hz	10.0 amps
240 vac, 60 Hz	7.5 amps

Contact Ratings, UL – Series EGP/EML Only
Contact ratings as listed under the Underwriters Laboratory Component Recognition Program. (Two poles per load):
1/3 Horsepower, 120 vac
10 amps, General Purpose, 240 vac
120 vdc, 1.0 amp

Mechanical Life – Series EGP/EML/ETR
25,000 mechanical operations

Approximate Weight – Series EGP/EML/ETR
1 lb.

Transient Protection – Series ETR Only
A 1500 volt transient of less than 100 microseconds, or 1000 volts of less than 1 millisecond will not affect timing accuracy.

Timing Adjustment – Series ETR Only
Internal Fixed
Internal Potentiometer

Time Ranges – Series ETR Only

.15 to 3 Sec.	4 to 120 Sec.
.55 to 15 Sec.	10 to 300 Sec.
1 to 30 Sec.	2 to 60 Min.
2 to 60 Sec.	1 to 30 Min.

Repeat Accuracy – Series ETR Only
The repeat accuracy deviation (A_R) of a time-delay relay is a measure of the maximum deviation in the time-delay that will be experienced in five successive operations at any particular time setting of the relay and over the operating voltage and temperature range specified. Repeat accuracy is obtained from the following formula:

$$A_R = \pm 100 \frac{(T_1 - T_2)}{(T_1 + T_2)}$$

Where —
T_1 = Maximum Time Delay.
T_2 = Minimum Time Delay.

Replacement Schedule – Series EGP/EML/ETR
The qualified life of these relays is 25,000 electrical operations or 10 years from the date of manufacture, whichever occurs first.

The date of manufacture can be found in the first four (4) digits of the serial number on the nameplate

First two digits indicate the year. — XX

Second two digits indicate the week. — XX

Example
In the date code "7814" below:
"78" indicates the year 1978;
"14" indicates the 14th week (or April 3 through april 7).

Model	
Coil	125 VDC
Serial	78140028

Note
Tyco Electronics Corporation does not recommend the use of its products in the containment areas of Nuclear Power Generating Stations.

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AGASTAT

Nuclear Qualified Control Relays

OPERATING CHARACTERISTICS

Environmental Conditons (Qualified Life) — Series EGP/EML/ETR

Parameter	Min.	Normal	Max.
Temperature (°F)	40	70-104	156
Humidity (R.H. %)	10	40-60	95
Pressure	–	Atmospheric	—
Radiation (rads)	–	–	2.0 x 10^8 (Gamma)

Operating Conditions, Normal Environment – Series EGP/EML/ETR

Normal Operating Specifications	With DC Coils			With AC Coils	
	EGP	EML	ETR	EGP	ETR
Coil Operating Voltage, Nominal (rated)*	As Spec.	As Spec.	As Spec.	As Spec.	As Spec.
Pull-in (% of rated value)	80% Min.	85% Min.	80% Min.	85% Min.	85% Min.
Drop-out (% of rated value)	5-45%	85% Min.	5-45%	5-45%	5-50%
Continuous (% of rated value)	110% Max.	N/A	110% Max.	110% Max.	110% Max.
Power (Watts at rated value)					
Pull-in	6 Apprx.	15 Apprx.	6 Apprx.	6 Apprx.	6 Apprx.
Drop-out	N/A	13 Apprx.	N/A	N/A	N/A
Relay Operate Time	30 ms Max.	25 ms Max. With min. latch pulse of 30 ms.	N/A	35 ms Max.	N/A
Relay Release (Recycle) Time	25 ms Max.	20 ms Max. With min. latch pulse of 30 ms.	75 ms Max.	85 ms Max.	75 ms Max.
Contact Ratings, Continuous					
Resistive at 125 vdc	1.0 amp.	1.0 amp.	1.0 amp.	1.0 amp.	1.0 amp.
Resistive at 120 vac, 60 Hz	10.0 amp.	10.0 amp.	10.0 amp.	10.0 amp.	10.0 amp.
Insulation Resistance (in megohms at 500 vdc)	500 Min.	500 Min.	500 Min.	500 Min.	500 Min.
Dielectric (vrms, 60 Hz)					
Between Terminals and Ground	1,500	1,500	1,500	1,500	1,500
Between Non-connected Terminals	1,500	1,500	1,500	1,500	1,500
Repeat Accuracy	N/A	N/A	±5%	N/A	±5%

Operating Conditions, Abnormal Environment – Series EGP/EML

Adverse Operating Specifications	Normal	DB "A"	DB "B"	DB "C"	DB "D"
Temperature (°F)	70-104	40	120	145	156
Humidity (R.H. %)	40-60	10-95	10-95	10-95	10-95
Coil Operating Voltage (% of rated)*					
AC (Series EGP only)	85-110	85-110	85-110	85-110	85-110
DC (Series EGP only)	80-110	80-110	80-110	80-110	80-110
DC (Series EML only)	85-110	85-110	85-110	85-110	85-110
Relay Operate Time (ms)					
AC (Series EGP only)	35 Max.	35 Max.	35 Max.	35 Max.	35 Max.
DC (Series EGP, Series EML)	30 Max.	25 Max.	37 Max.	40 Max.	40 Max.

Operating Conditions, Abnormal Environment – Series ETR

Adverse Operating Specifications	With DC Coils	With AC Coils
Coil Operating Voltage (rated)*	As Spec.	As Spec.
Pull-in (% of rated value)	80% Min.	85% Min.
Continuous (% of rated value)	110% Max.	110% Max.
Drop-out (% of rated value)	5-45%	5-50%
Power (Watts at rated value)	6 Apprx.	6 Apprx.
Relay Release (Recycle) Time	75 ms Max.	75 ms Max.
Contact Ratings, Continuous		
Resistive at 125 vdc	1.0 amp.	1.0 amp.
Resistive at 120 vac, 60 Hz	10.0 amp.	10.0 amp.
Repeat Accuracy	±10%	±10%

*All coils may be operated on intermittent duty cycles at voltages 10% above listed maximums (Intermittent Duty = Maximum 50% duty cycle and 30 minutes "ON" time.)

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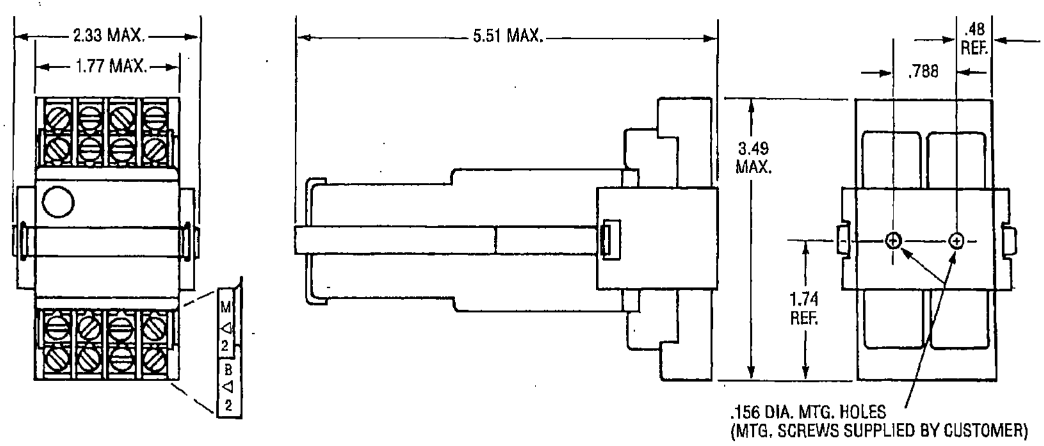
AGASTAT

Nuclear Qualified Control Relays

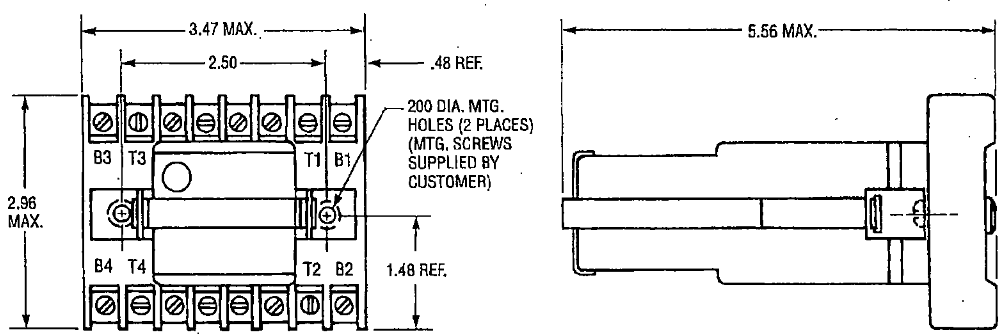
DIMENSIONS AND MOUNTING

SERIES EGP, EML AND ETR

All dimensions in inches



Qualification tested in the horizontal position, mounted in socket ECR0001-001 (captive clamp terminals) or in socket ECR0002-001 (screw terminals) with locking straps ECR0133.



Qualification tested in the horizontal position, mounted in socket ECR0095-001 (screw terminals) with locking strap ECR0155.

Series EGP, EML and ETR AGASTAT® control *relays must be mounted in the horizontal position*; performance specifications of these units are valid only when they are mounted as indicated in either of the above drawings.

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AGASTAT®

Nuclear Qualified Control Relays

ORDERING INFORMATION

Catalog Number Code – Series EGP and EML

E	GP	A	004
Nuclear Safety Related	AGASTAT® Control Relay Model	Coil Voltage	Configuration Code*
Code E	Code GP – Power Relay ML – Magnetic Latch	Code DC: A – 12 VDC B – 24 VDC C – 48 VDC D – 125 VDC E – 110 VDC F – 250 VDC (Series EGP Only) AC: G – 24 VAC 60 Hz (Series EGP Only) H – 48 VAC 60 Hz (Series EGP Only) I – 120 VAC 60 Hz (Series EGP Only) J – 220 VAC 60 Hz (Series EGP Only)	Code 004

* Configuration Code
The Configuration Code is a suffix to the Model Number which provides a means of identification. When a significant product change is introduced, the Configuration code and specification sheets will be revised.

E	TR14	B	1	A	**004
Nuclear Safety Related	AGASTAT® Control Relay Model	Operating Voltage	Timing Adjustment	Time Range	Configuration Code
Code E	Code TR14 – Time Delay Relay (Delay on Pull-in)	Code DC: B – 24 VDC D – 125 VDC AC: I – 120 VAC 60Hz	Code 1 – Internal Fixed 3 – Internal Potentiometer	Code A – .15 to 3 sec. B – .55 to 15 sec. C – 1 to 30 sec. D – 2 to 60 sec. E – 4 to 120 sec. G – 10 to 300 sec. I – 2 to 60 min. N – 1 to 30 min.	Code 004

* Configuration Code
The Configuration Code is a suffix to the Model Number which provides a means of identification. When a significant product change is introduced, the Configuration code and specification sheets will be revised.

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AGASTAT

Relay Classifications Control Code Summary

CONFIGURATION CONTROL

Product	Code – 001	Code – 002	Code – 003	Code – 004
EGP	Contains all materials present in original qualification testing.	Nov. 1981 - Material change to coil wrapping tape and lead wire insulation to improve thermal life.	Dec. 1987 - Material change on leaf spring from nickel copper to beryllium copper.	Dec. 1995 - Material change on bobbin from Nylon Zytel 101 to Rynite FR530. Material change on base from Melamine Phenolic to Grilon PMV-5HV0.
EML	Contains all materials present in original qualification testing.	Nov. 1981 - Material change to coil wrapping tape and lead wire insulation to improve thermal life.	Dec. 1987 - Material change on leaf spring from nickel copper to beryllium copper.	Dec. 1995 - Material change on bobbin from Nylon Zytel 101 to Rynite FR530. Material change on base from Melamine Phenolic to Grilon PMV-5HV0.
ETR	Contains all materials present in original qualification testing.	Nov. 1981 - Material change to coil wrapping tape and lead wire insulation to improve thermal life.	Dec. 1987 - Material change on leaf spring from nickel copper to beryllium copper.	Dec. 1995 - Material change on bobbin from Nylon Zytel 101 to Rynite FR530. Material change on base from Melamine Phenolic to Grilon PMV-5HV0.
ECR0001	Contains all materials present in original qualification testing.	June 1989 - Material change from Noryl N-225 std. black to Noryl SE-I-701AA black.		
ECR0002	Contains all materials present in original qualification testing.	June 1989 - Material change from Noryl N-225 std. black to Noryl SE-I-701AA black.		
ECR0095	Contains all materials present in original qualification testing.			
ECR0133	Contains all materials present in original qualification testing.			
ECR0155	Contains all materials present in original qualification testing.			

Configuration Code: The Configuration code is a suffix to the Model Number which provides a means of identification. When a significant product change is introduced, the Configuration code and specification sheets will be revised. (001, 002, 003, 004, etc.).

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ATT. B.
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 7, 2012

Mr. Mark A. Schimmel
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT REGARDING THE AUTOMATIC DEPRESSURIZATION SYSTEM BYPASS TIMER (TAC NO. ME8345)

Dear Schimmel:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 170 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the technical specifications (TSs) in response to your application dated April 5, 2012.

The amendment revises TSs to eliminate the lower allowable value limit of "≥ 18 minutes" for Functions 1.e and 2.e, "Reactor Steam Dome Pressure Permissive – Bypass Timer (Pump Permissive)," in Table 3.3.5.1-1, "Emergency Core Cooling System Instrumentation."

Please note that the NRC staff declined to issue this amendment by the licensee's requested issuance date of June 6, 2012; the application did not provide any reason for shortening the regulatory 60-day notice period during which interested parties may petition for a hearing.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 170 to DPR-22
2. Safety Evaluation

cc w/encls: Distribution via ListServ