# ENCLOSURE 2

# GE-MNGP-AEP-1913 R1

# GEH Responses to Reactor Systems RAIs – Non-proprietary

**NON-PROPRIETARY NOTICE**

This is a non-proprietary version of the Enclosure 1 of GE-MNGP-AEP-1913 R1 which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

**Reactor Systems RAI-01**

*Section 2.1.1 of NEDC-33435P/Rev1 states that "no additional fuel and core design evaluation is required" because Monticello will use a full load of GE14 for the first MELLLA+ core reload. What fuel evaluations will be required for future core reloads if fuel other than GE14 is used?*

**GEH Response**:

The implementation of a new fuel design from Global Nuclear Fuel, LLC (GNF) into a GE Hitachi Nuclear Energy (GEH) Boiling Water Reactor (BWR) follows a two-step process. First, the new fuel design is approved by the Nuclear Regulatory Commission generically via the GESTAR II Amendment 22 fuel compliance process. Then, plant-specific analyses are performed to justify use of the new fuel design in the plant reload. The plant-specific analyses consist of one-time cycle-independent analyses and normal cycle-dependent analyses. The cycle-dependent analyses (e.g., cold shut-down reactivity margin through the cycle, core stability performance, margin to the vessel over-pressure safety limit) are necessary for each reload regardless of fuel design.

For a GEH BWR such as Monticello, the cycle-independent work required to introduce a new GNF fuel design consists of the tasks listed below. Sufficient evaluation for each topic is required to demonstrate that current analyses are applicable or to update such analyses.

- Evaluation of the stability solution planned for use with the new fuel design,
- Evaluation of the effect of the new fuel design on the decay heat used for plant analyses such as containment,
- Analysis of Loss-of-Coolant Accident (LOCA),
- Calculation of Reactor Internal Pressure Differences (RIPD) for Normal, Upset, Faulted, and Emergency conditions,
- Calculation of Seismic loads affected by the fuel design change,
- Evaluation of the RIPD and Seismic load changes on affected reactor internal components,
- Analysis of the reactor recirculation pump seizure event during single-loop operation,
- Calculation of certain off-rated power and flow dependent limits not specifically addressed by cycle-dependent analysis,
- Evaluation of the limiting Appendix R event,
- Evaluation of the plant response to an Anticipated Transient Without Scram (ATWS),
- Evaluation of the effect that the fuel design change has on the neutron fluence,
- Evaluation of the Fuel Handling Accident (FHA),

- Evaluation of the effect that the new fuel design has on the source term,
- Calculation of Emergency Operating Procedure parameters,
- Evaluation of mechanical compatibility of the new fuel design with core components and fuel storage and handling equipment,
- Analysis of the margin to fuel storage criticality limits, and
- Assessment of the effect of the new fuel design on the performance of the recirculation system.

Consistent with the normal cycle-dependent analyses and as appropriate, these fuel design dependent, cycle-independent evaluations, analyses, calculations, and assessments consider the entire allowable operating domain for a BWR, which would include MELLLA+ for those plants licensed for MELLLA+.

**Reactor Systems RAI-02**

*Section 2.1.2 of NEDC-33435P/Rev1 states that "Because there is no increase in the average bundle power or in the maximum allowable peak bundle power there is no change required to the fuel thermal monitoring threshold." Figures 2-7 through 2-17 provide 2D bundle distributions for relevant parameters for the representative MELLLA+ core. Provide similar figures for the last non-MELLLA+ Monticello core.*

**GEH Response:**

As clarified and agreed on the phone call with the NRC on July 19, 2010, the following figures for the cycle 25 Extended Power Uprate (EPU) reload licensing core are "representative" of the last non-MELLLA+ Monticello core. The ¼ core maps are shown as Figures RAI-02 2-7 through 2-17 consistent with the figures (and figure numbers) in NEDC-33435P, Revision 1 (Reference RAI-02-1). The cycle 25 licensing calculations were performed at an EPU power of 2004 MWt and designed for nominal cycle energy of [[ .............................................. ]]. The 200 MWd/ST cycle exposure represents the beginning-of-cycle (BOC) point, the 8,500 MWd/ST cycle exposure represents the middle-of-cycle (MOC) point, and the 12,700 MWd/ST cycle exposure represents the end-of-cycle (EOC) point. The peak Maximum Fraction of Limiting Critical Power Ratio (MFLCPR) point is at a cycle exposure of 11,000 MWd/ST and the peak Maximum Fraction of Limiting Power Density (MFLPD) point is at a cycle exposure of 12,500 MWd/ST. All points are adjusted to be at the 100% EPU power level. The flows at the BOC, MOC, peak MFLCPR, and MFLPD points are adjusted to the 99% rated flow and the EOC point is adjusted to the 105% rated flow. The EOC point is an all rods out condition.

**Reference:**

RAI-02-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

**Figure RAI-02 2.7 Dimensionless Bundle Power at 200 MWd/ST**

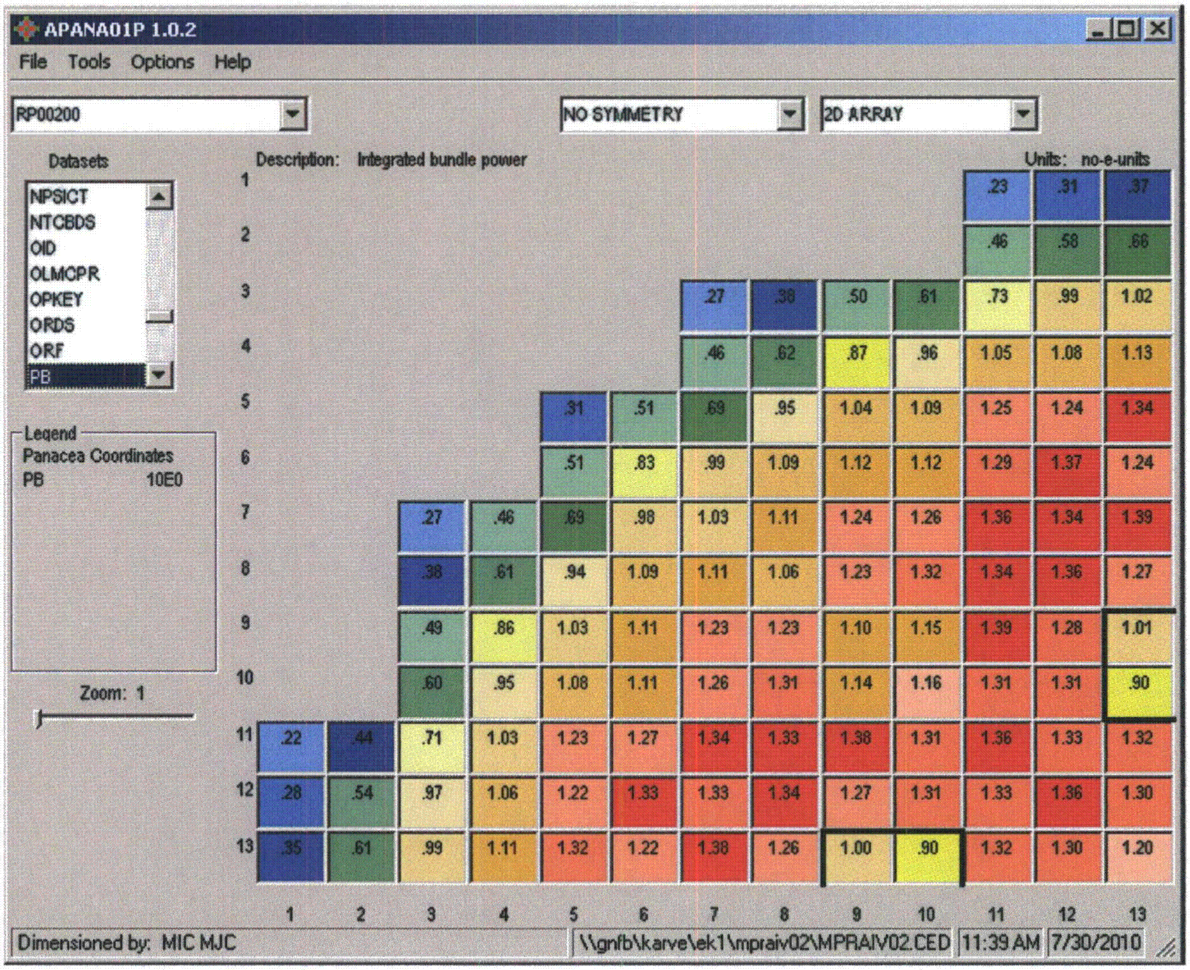


**Figure RAI-02 2.8 Dimensionless Bundle Power at 8500 MWd/ST**

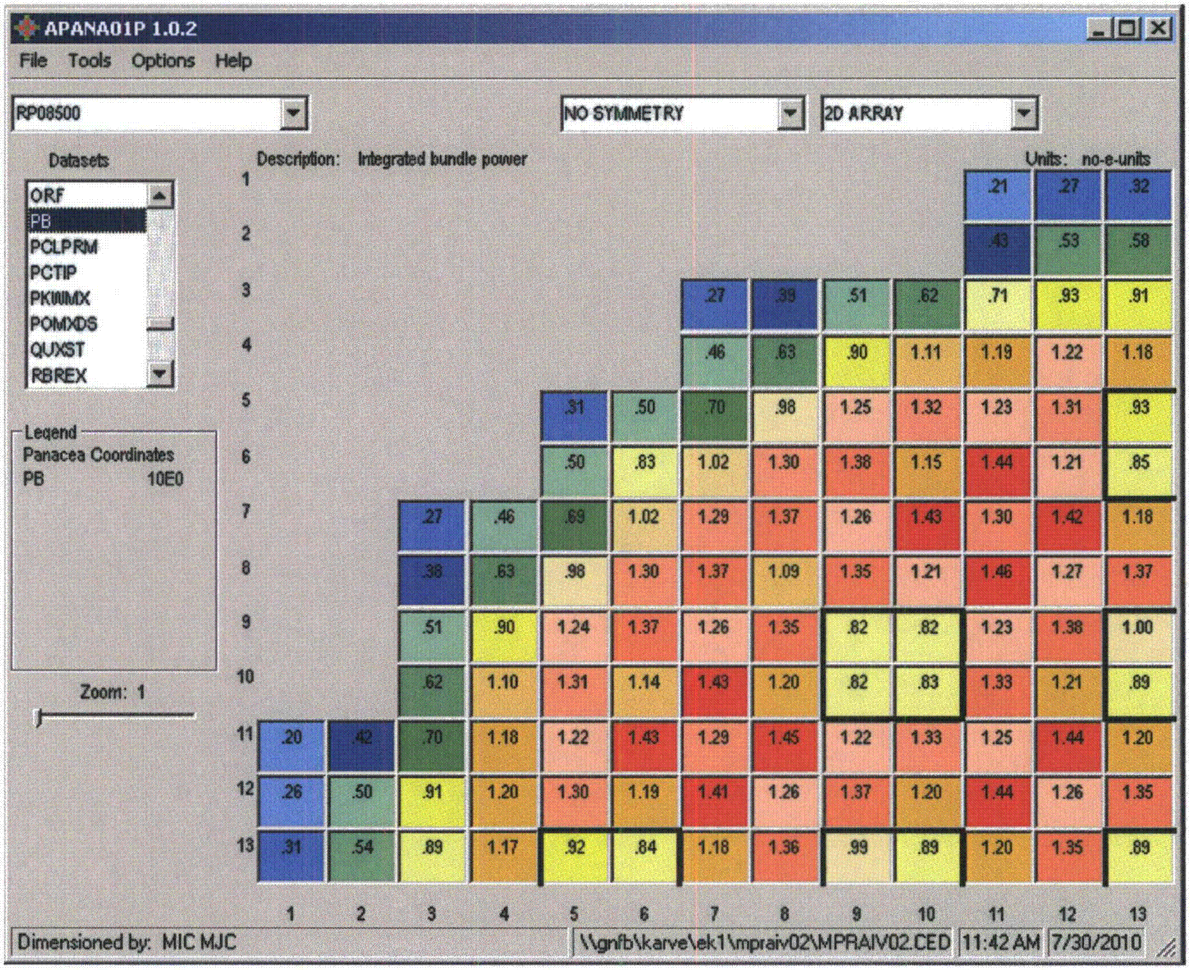


**Figure RAI-02 2.9 Dimensionless Bundle Power at 12,700 MWd/ST**

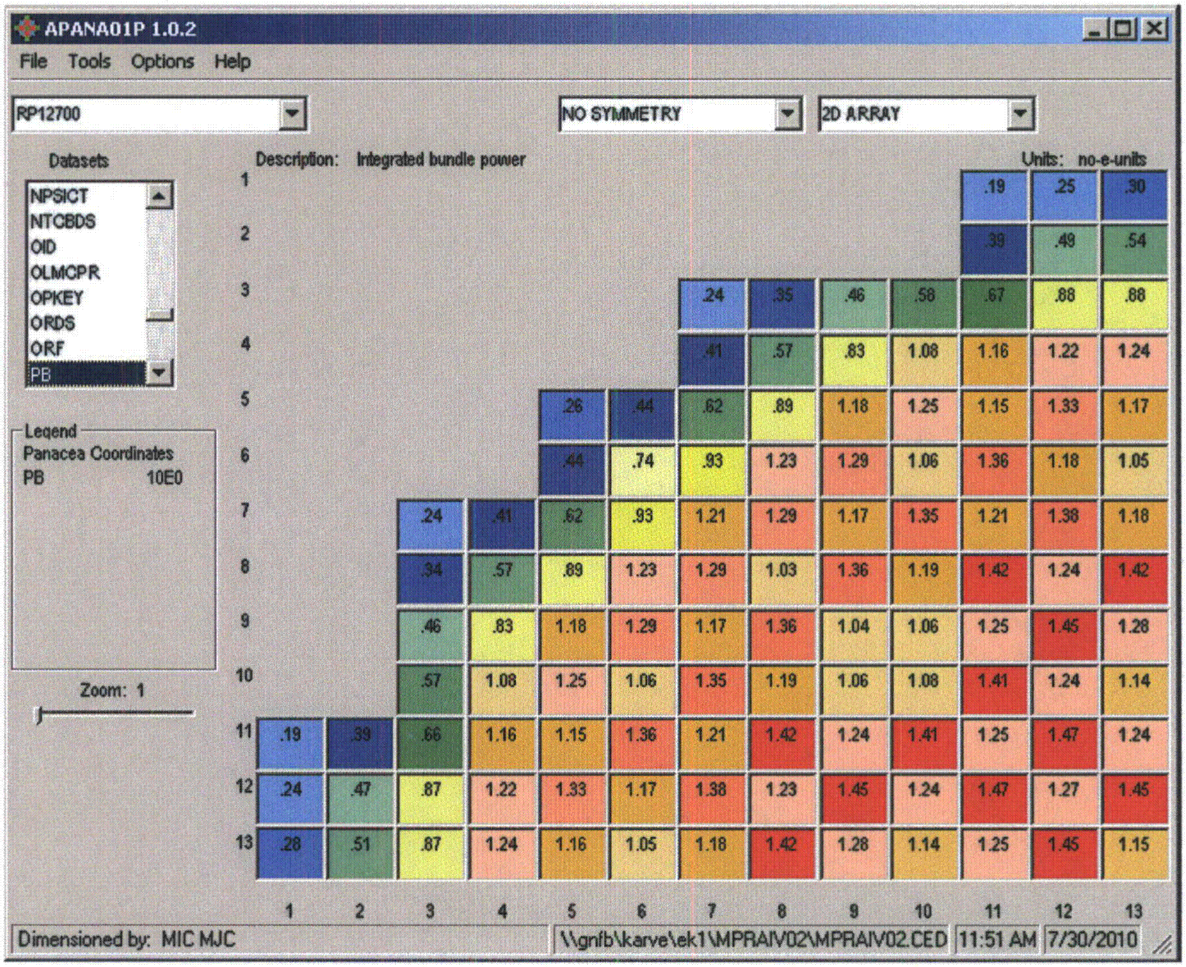
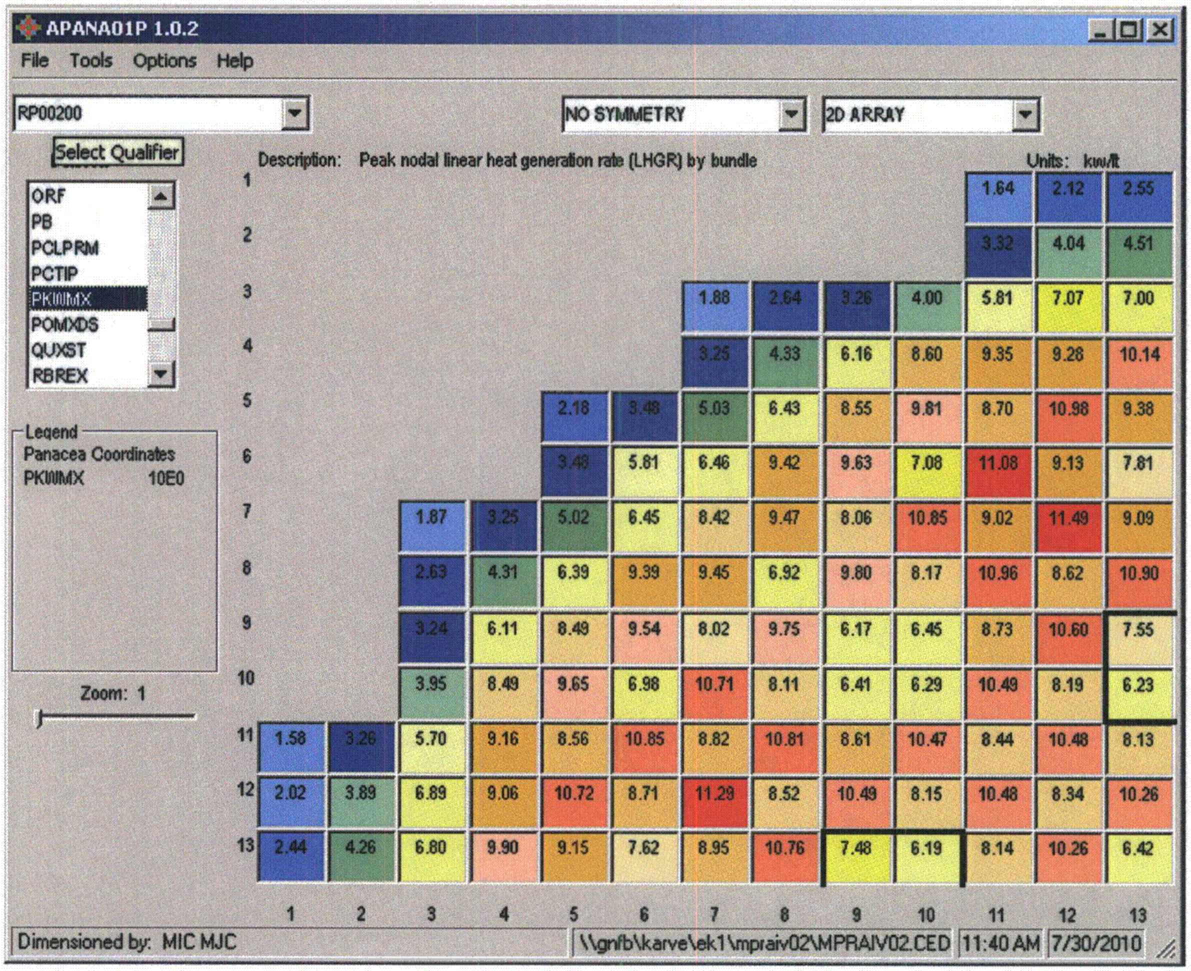
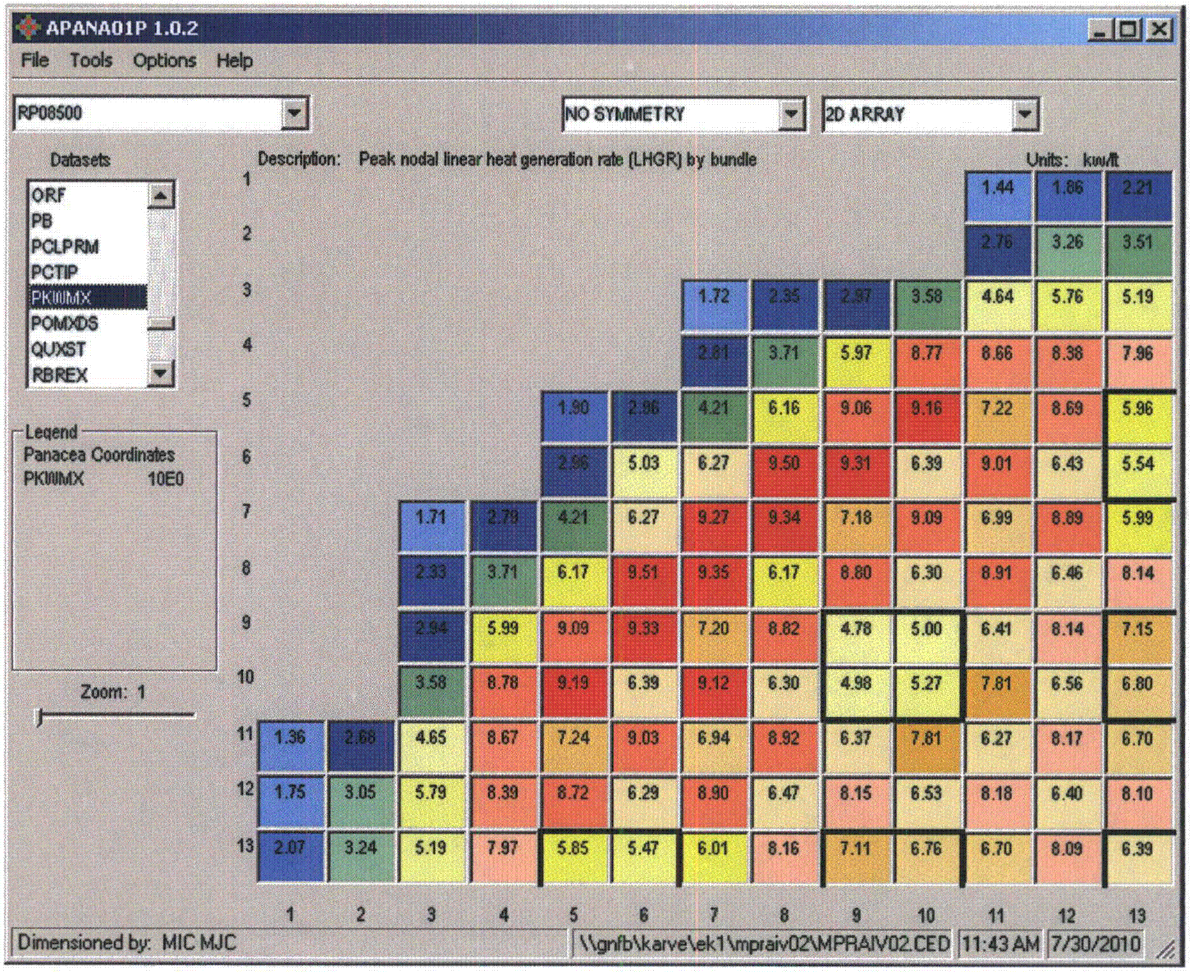


**Figure RAI-02 2.10 Bundle Operating LHGR (kW/ft) at 200 MWd/ST**




| | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | | | | | | | | | | | 1.64 | 2.12 | 2.55 |
| 2 | | | | | | | | | | | 3.32 | 4.04 | 4.51 |
| 3 | | | | | | | 1.88 | 2.64 | 3.26 | 4.00 | 5.81 | 7.07 | 7.00 |
| 4 | | | | | | | 3.25 | 4.33 | 6.16 | 8.60 | 9.35 | 9.28 | 10.14 |
| 5 | | | | | 2.18 | 3.48 | 5.03 | 6.43 | 8.55 | 9.81 | 8.70 | 10.98 | 9.38 |
| 6 | | | | | 3.48 | 5.81 | 6.46 | 9.42 | 9.63 | 7.08 | 11.08 | 9.13 | 7.81 |
| 7 | | | 1.87 | 3.25 | 5.02 | 6.45 | 8.42 | 9.47 | 8.06 | 10.85 | 9.02 | 11.49 | 9.09 |
| 8 | | | 2.63 | 4.31 | 6.39 | 9.39 | 9.45 | 6.92 | 9.80 | 8.17 | 10.96 | 8.62 | 10.90 |
| 9 | | | 3.24 | 6.11 | 8.49 | 9.54 | 8.02 | 9.75 | 6.17 | 6.45 | 8.73 | 10.60 | 7.55 |
| 10 | | | 3.95 | 8.49 | 9.65 | 6.98 | 10.71 | 8.11 | 6.41 | 6.29 | 10.49 | 8.19 | 6.23 |
| 11 | 1.58 | 3.26 | 5.70 | 9.16 | 8.56 | 10.85 | 8.82 | 10.81 | 8.61 | 10.47 | 8.44 | 10.48 | 8.13 |
| 12 | 2.02 | 3.89 | 6.89 | 9.06 | 10.72 | 8.71 | 11.29 | 8.52 | 10.49 | 8.15 | 10.48 | 8.34 | 10.26 |
| 13 | 2.44 | 4.26 | 6.80 | 9.90 | 9.15 | 7.62 | 8.95 | 10.76 | 7.48 | 6.19 | 8.14 | 10.26 | 6.42 |

**Figure RAI-02 2.11 Bundle Operating LHGR (kW/ft) at 8500 MWd/ST**




| | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | | | | | | | | | | | 1.44 | 1.86 | 2.21 |
| 2 | | | | | | | | | | | 2.76 | 3.26 | 3.51 |
| 3 | | | | | | | 1.72 | 2.35 | 2.97 | 3.58 | 4.64 | 5.76 | 5.19 |
| 4 | | | | | | | 2.81 | 3.71 | 5.97 | 8.77 | 8.66 | 8.38 | 7.96 |
| 5 | | | | | 1.90 | 2.96 | 4.21 | 6.16 | 9.06 | 9.16 | 7.22 | 8.69 | 5.96 |
| 6 | | | | | 2.96 | 5.03 | 6.27 | 9.50 | 9.31 | 6.39 | 9.01 | 6.43 | 5.54 |
| 7 | | | 1.71 | 2.79 | 4.21 | 6.27 | 9.27 | 9.34 | 7.18 | 9.09 | 6.99 | 8.89 | 5.99 |
| 8 | | | 2.33 | 3.71 | 6.17 | 9.51 | 9.35 | 6.17 | 8.80 | 6.30 | 8.91 | 6.46 | 8.14 |
| 9 | | | 2.94 | 5.99 | 9.09 | 9.33 | 7.20 | 8.82 | 4.78 | 5.00 | 6.41 | 8.14 | 7.15 |
| 10 | | | 3.58 | 8.78 | 9.19 | 6.39 | 9.12 | 6.30 | 4.98 | 5.27 | 7.81 | 6.56 | 6.80 |
| 11 | 1.36 | 2.68 | 4.65 | 8.67 | 7.24 | 9.03 | 6.94 | 8.92 | 6.37 | 7.81 | 6.27 | 8.17 | 6.70 |
| 12 | 1.75 | 3.05 | 5.79 | 8.39 | 8.72 | 6.29 | 8.90 | 6.47 | 8.15 | 6.53 | 8.18 | 6.40 | 8.10 |
| 13 | 2.07 | 3.24 | 5.19 | 7.97 | 5.85 | 5.47 | 6.01 | 8.16 | 7.11 | 6.76 | 6.70 | 8.09 | 6.39 |

**Figure RAI-02 2.12 Bundle Operating LHGR (kW/ft) at 12,700 MWd/ST**

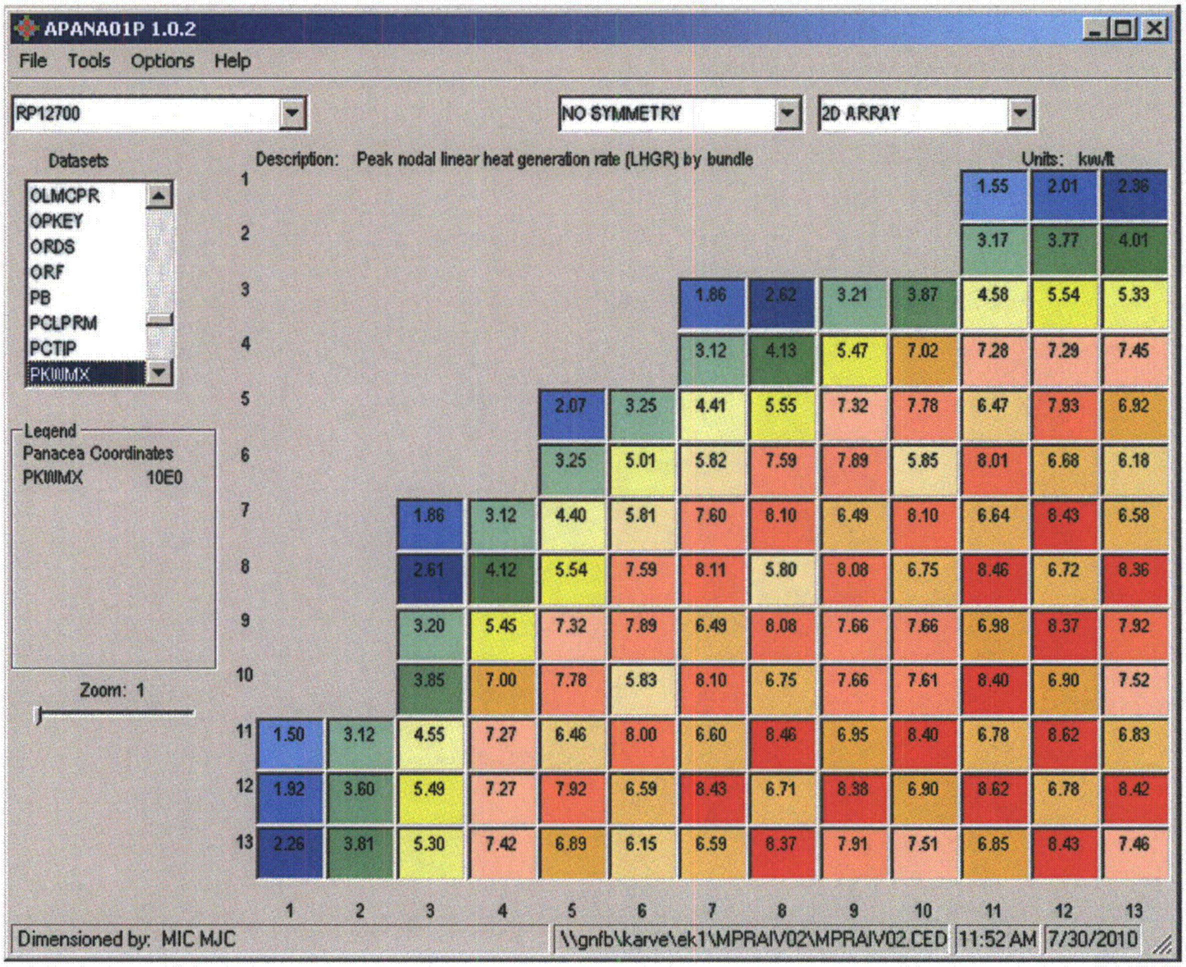


**Figure RAI-02 2.13 Bundle operating MCPR at 200 MWd/ST**

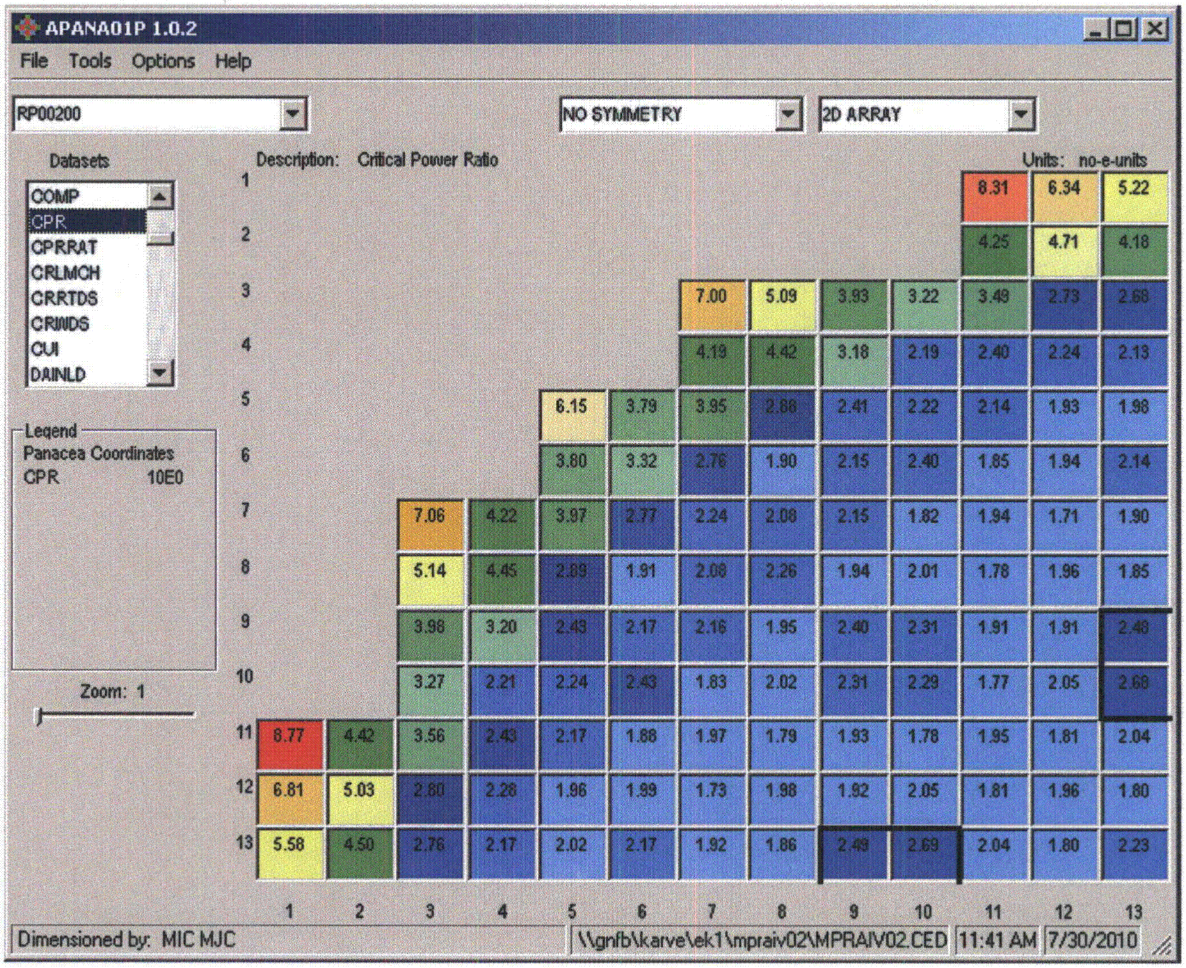


**Figure RAI-02 2.14 Bundle operating MCPR at 8500 MWd/ST**

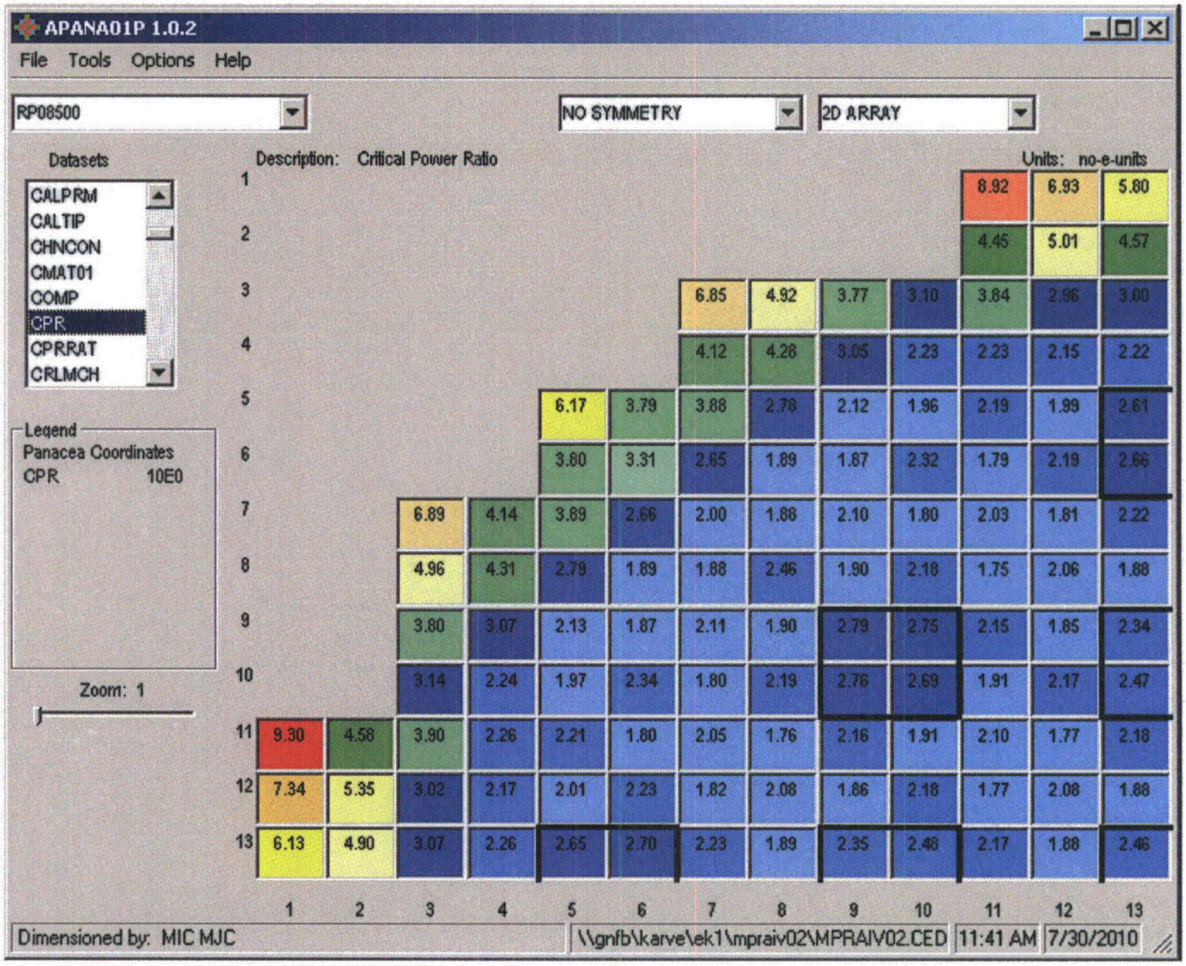


**Figure RAI-02 2.15 Bundle operating MCPR at 12,700 MWd/ST**

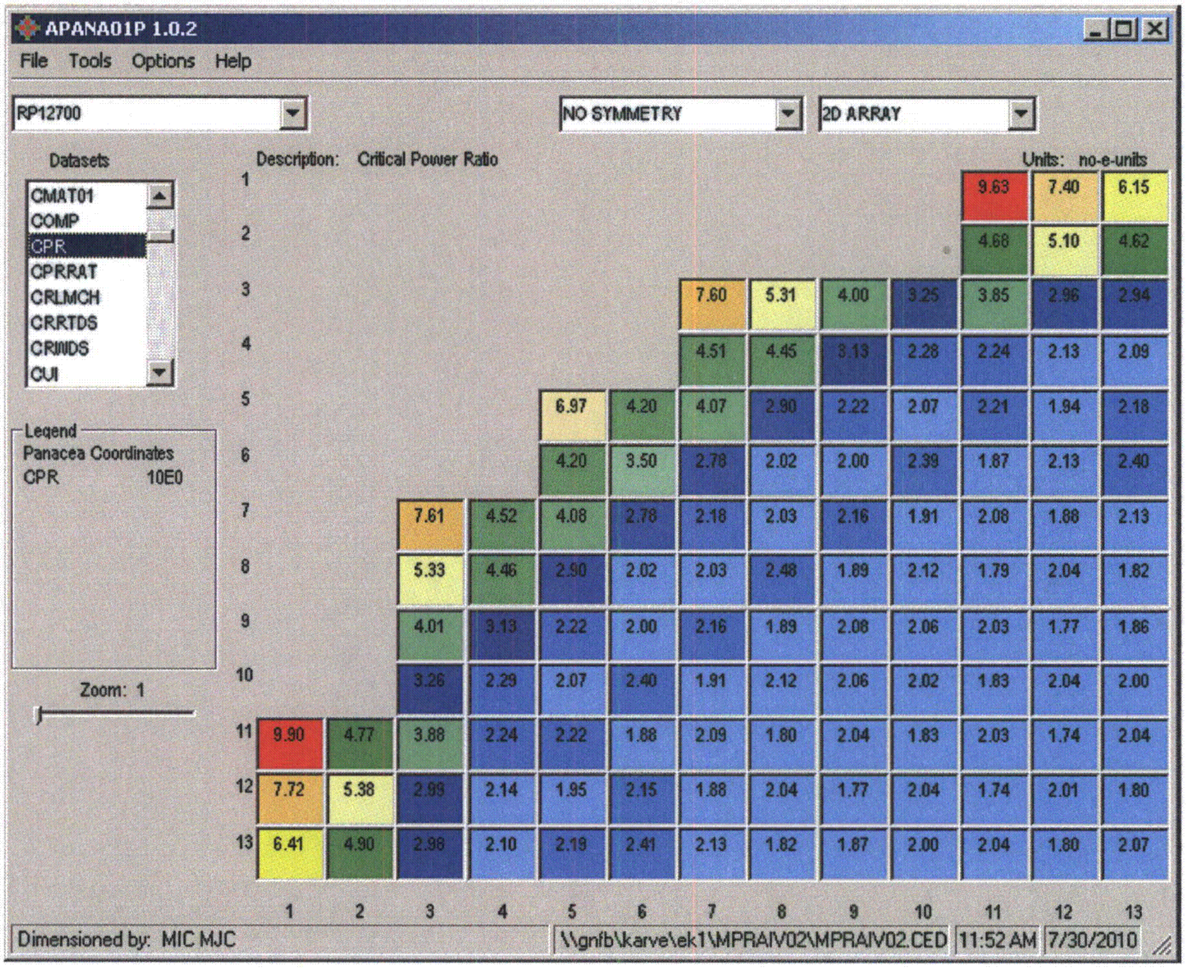
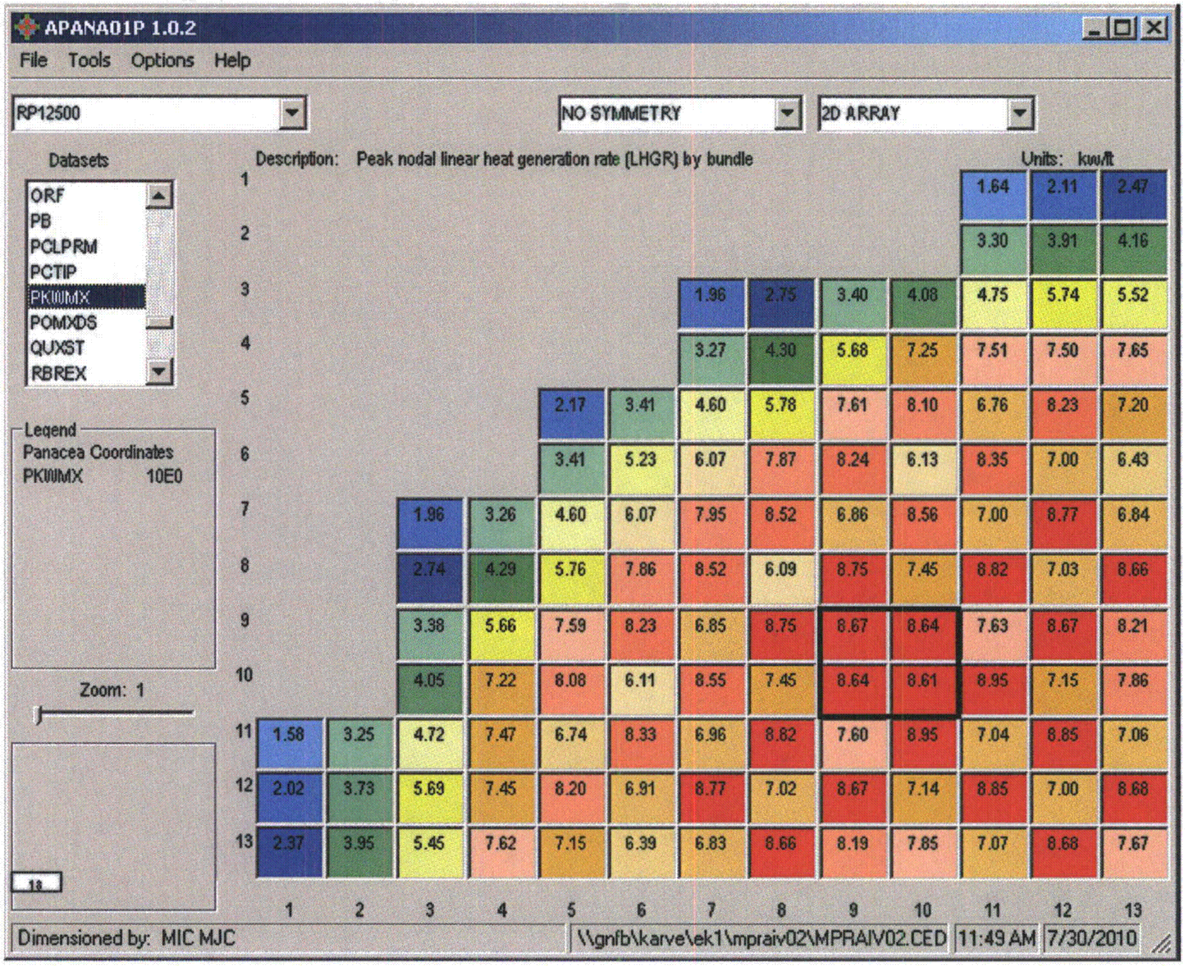


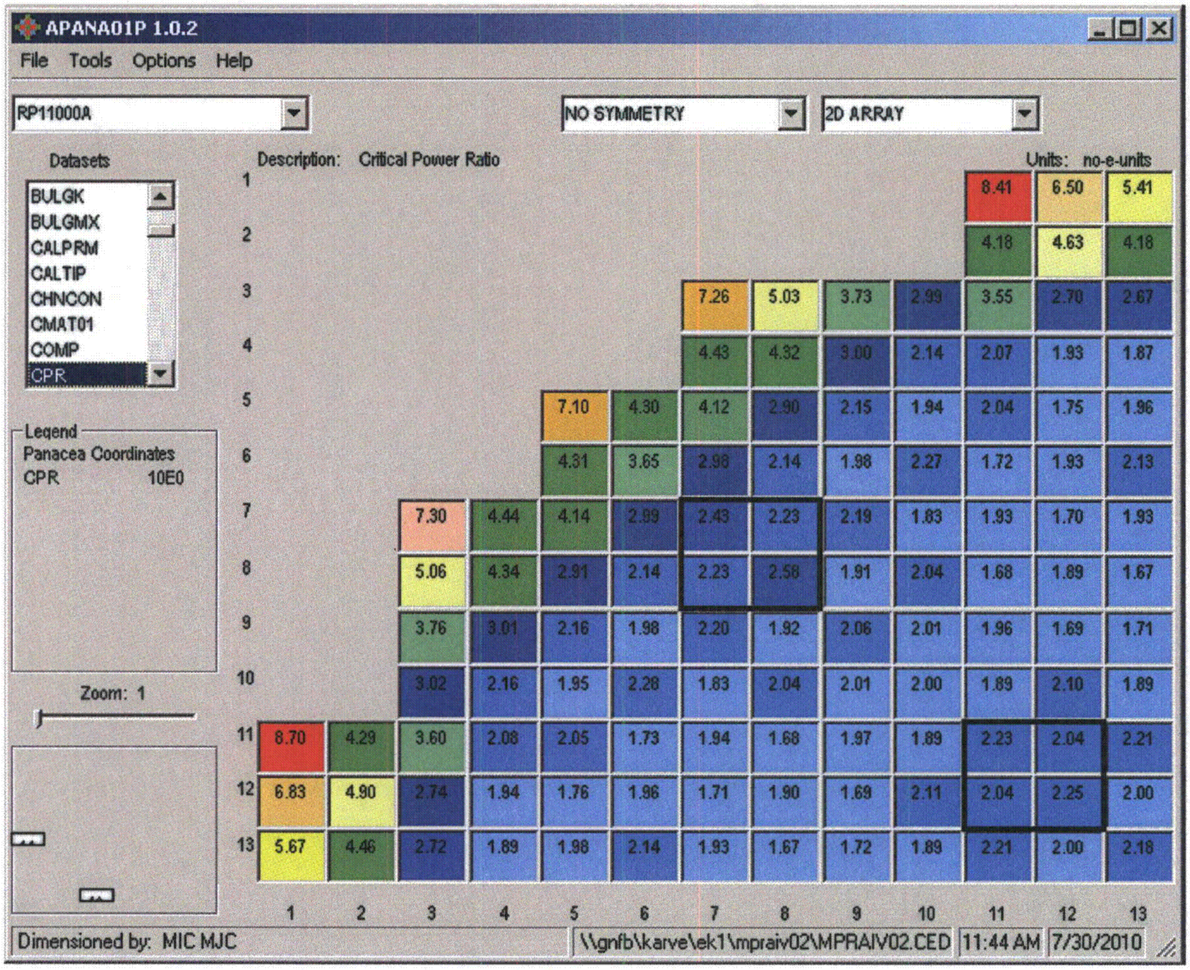
**Figure RAI-02 2.16 Bundle Operating LHGR (kW/ft) at 12500 MWd/ST (peak MFLPD*)**




| | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | | | | | | | | | | | 1.64 | 2.11 | 2.47 |
| 2 | | | | | | | | | | | 3.30 | 3.91 | 4.16 |
| 3 | | | | | | | 1.96 | 2.75 | 3.40 | 4.08 | 4.75 | 5.74 | 5.52 |
| 4 | | | | | | | 3.27 | 4.30 | 5.68 | 7.25 | 7.51 | 7.50 | 7.65 |
| 5 | | | | | 2.17 | 3.41 | 4.60 | 5.78 | 7.61 | 8.10 | 6.76 | 8.23 | 7.20 |
| 6 | | | | | 3.41 | 5.23 | 6.07 | 7.87 | 8.24 | 6.13 | 8.35 | 7.00 | 6.43 |
| 7 | | | 1.96 | 3.26 | 4.60 | 6.07 | 7.95 | 8.52 | 6.86 | 8.56 | 7.00 | 8.77 | 6.84 |
| 8 | | | 2.74 | 4.29 | 5.76 | 7.86 | 8.52 | 6.09 | 8.75 | 7.45 | 8.82 | 7.03 | 8.66 |
| 9 | | | 3.38 | 5.66 | 7.59 | 8.23 | 6.85 | 8.75 | 8.67 | 8.64 | 7.63 | 8.67 | 8.21 |
| 10 | | | 4.05 | 7.22 | 8.08 | 6.11 | 8.55 | 7.45 | 8.64 | 8.61 | 8.95 | 7.15 | 7.86 |
| 11 | 1.58 | 3.25 | 4.72 | 7.47 | 6.74 | 8.33 | 6.96 | 8.82 | 7.60 | 8.95 | 7.04 | 8.85 | 7.06 |
| 12 | 2.02 | 3.73 | 5.69 | 7.45 | 8.20 | 6.91 | 8.77 | 7.02 | 8.67 | 7.14 | 8.85 | 7.00 | 8.68 |
| 13 | 2.37 | 3.95 | 5.45 | 7.62 | 7.15 | 6.39 | 6.83 | 8.66 | 8.19 | 7.85 | 7.07 | 8.68 | 7.67 |

* Maximum Fraction of Limiting Power Density

**Figure RAI-02 2.17 Bundle operating MCPR at 11000 MWd/ST (peak MFLCPR* point)**




* Maximum Fraction of Limiting Critical Power Ratio

**Reactor Systems RAI-03**

*Section 2.1.2 of NEDC-33435P/Rev1 stats that "For Monticello, the predicted bypass void fraction at the D-Level Local Power Range Monitor (LPRM) is less than the [[ ]] design requirement". Identify the methodology used to perform the reported bypass void analysis (i.e., ISCOR hot channel, ISCOR average channel, TRACG ...). Provide the results of the bypass void analysis and identify the limiting operating conditions assumed.*

**GEH Response**:

The methodology used is ISCOR hot channel. The bypass void at the D-level LPRM is equal to [[ ]]. This is evaluated at the limiting operating condition of the lower elbow point on the MELLLA+ upper boundary, i.e., the [[ ]] rated power and the [[ ]] rated flow point.

*(2) Provide a description or a reference to justify the TRACG reactivity biases used.*

The TRACG nodal void reactivity model used in the TRACG ATWS with Core Instability analysis is based on the default model available in TRACG (i.e., the PIRT18 response surface for void coefficient reactivity was used). This is consistent with NEDE-32906P-A, Revision 3 (Reference RAI-04-2) and is the approach used for TRACG AOO and Stability analyses.

The TRACG nodal void reactivity model used in the TRACG ATWS with Depressurization analyses is based on the alternate model available in TRACG (i.e., the PIRT18 response surface for void coefficient reactivity was not used). In accordance with Limitation and Condition 12.18.a of NEDC-33006P-A, Revision 3 (Reference RAI-04-3) the TRACG ATWS with Depressurization analyses were performed as alternate demonstration analyses to take advantage of the TRACG capability to model the Monticello Emergency Operating Procedure (EOP) actions such as emergency depressurization. Variation in operator mitigation strategy (water level control), Heat Capacity Temperature Limit (HCTL) temperatures and core exposure were investigated. The nodal void reactivity model used for these simulations was considered adequate for this purpose. These analyses demonstrated that the ODYN licensing basis analyses are bounding.

**References:**

RAI-04-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

RAI-04-2 NEDE-32906P-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," September 2006

RAI-04-3 NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," June 2009

**Reactor Systems RAI-05**

*Section 2.3.3 of NEDC-33435P/Rev1 states that the SLCS shutdown margin is evaluated to ensure it remains within Tech Specs. Provide the Hot Shutdown Boron Weight (HSBW) and its injection time for MELLLA+ and the last non-MELLLA+ core in Monticello.*

**GEH Response**:

The Technical Specification shutdown margin check is based on the cold shutdown condition. For Monticello, the Standby Liquid Control System (SLCS) injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor coolant at 68°F.

The hot shutdown boron weight (HSBW) is calculated generically for a fuel product line for use in Emergency Procedure Guideline (EPG) calculations to determine the amount of gallons required to achieve the hot shutdown condition. ATWS analyses are performed on a plant specific basis to confirm the adequacy of the boron injection rate. The HSBW calculation performed for EPGs is based on the following conditions:

(1) The reactor has been operating on the Maximum Extended Operating Domain load line.

(2) Control rods are withdrawn to the maximum rod block limit.

(3) The reactor core is at the most reactive exposure.

(4) Full power equilibrium xenon is present in the reactor core.

(5) No voids are present in the core.

(6) Reactor Pressure Vessel (RPV) pressure is 1100 psia.

(7) RPV water level is at the high level trip setpoint.

(8) No shutdown cooling is in service.

The combination of these conservative assumptions is used to arrive at the amount of gallons required to achieve the desired ppm when well mixed in the vessel. The calculated HSBW is about 460 gallons. The HSBW injection time for Monticello is 19.2 minutes. The injection time includes conservatism in the SLCS delivery rate compared to the design flow rate. This is the HSBW injection time for both MELLLA+ and non-MELLLA+ conditions. The ATWS analyses with both ODYN and TRACG confirm that the injection time is adequate to meet the requirements.

**Reactor Systems RAI-06**

*Section 2.4.1 of NEDC-33435P/Rev1 and tables 2-2 through 2-4 provide the conclusions to a series of TRACG analyses to demonstrate satisfactory DSS/CD application when the amplitude discrimination setpoint is increased from the generic value of 1.03 to as high as 1.10. Provide additional detail about the procedure used to perform these calculations and the methodology to ensure that the SAD value of 1.10 provides similar final MCPR margin than a value of 1.03.*

**GEH Response**:

Additional details about the procedure used to perform the calculations whose results are documented in Section 2.4.1 and in Tables 2-2 through 2-4 of NEDC-33435P, Revision 1 (Reference RAI-06-1) were provided in a presentation shared with NRC staff reviewers on May 4, 2010. [[

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The NRC staff reviewers acknowledged that the content of the presentation provides sufficient explanation on the procedure used to perform the aforementioned calculations and methodology. The NRC staff reviewers also mentioned that Monticello specific values and results should be used in the presentation material.

Therefore, the May presentation was updated to include Monticello MELLLA+/DSS-CD specific values and results from Reference RAI-06-1. The resulting updated presentation is included on the Enclosure 3 CD ROM with the following file name:

"001_DSS-CD_TRACG02_Application_Method_to_MNGP.pdf"

Note that this file is also referenced in the responses to RAI-08 and RAI-10.

**Reference:**

RAI-06-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

**Reactor Systems RAI-07**

*Provide the power and CPR time traces for the TRACG04 plant-specific demonstrations in Table 2-4 of NEDC-33435P/Rev1.*

**GEH Response**:

For each TRACG04 plant-specific demonstration case in Table 2-4 of NEDC-33435P, Revision 1 (Reference RAI-07-1), a plot of transient hot channel power and CPR time traces is provided in Figures RAI-07-1 through RAI-07-4. Please note that case 10080RG-TLO-1RPT did not result in oscillations as shown in Figure RAI-07-3.

**Reference:**

RAI-07-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

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**Reactor Systems RAI-08**

*Provide the actual values of final MCPR for the matrix cases in Table 2-3 of NEDC-33435P/Rev1. Provide the power and MCPR time traces for SAD=1.03 and SAD=1.10.*

**GEH Response**:

The values of Final MCPR (FMCPR) at the time of oscillation suppression for the matrix of cases in Table 2-3 of NEDC-33435P, Revision 1 (Reference RAI-08-1) are provided in Table RAI-08-1. The results are based on the TRACG02 confirmation cases documented in Section 4.0 of NEDC-33075P-A, Revision 6 (Reference RAI-08-2).

Additional analyses have been performed to determine the corresponding FMCPR with the increased Amplitude Discriminator Setpoint ($S_{AD}$) of 1.10. The results of these analyses are also provided in Table RAI-08-1. Please note that for all cases in Table RAI-08-1 the FMCPR is the bounding value that includes the DSS-CD prescribed MCPR uncertainty allowances from Reference RAI-08-2 [[

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[[

]] This process is explained in the updated presentation from the May 4, 2010 NRC-GEH meeting that is included on the Enclosure 3 CD ROM with the following file name:

"001_DSS-CD_TRACG02_Application_Method_to_MNGP.pdf"

Note that this file is also referenced in the responses to RAI-06 and RAI-10.

The plots of transient hot channel power and CPR traces are provided in Figures RAI-08-1 through RAI-08-7. This item was discussed with the NRC staff reviewers during the conference call held on July 19, 2010 to review the Draft RAI on Monticello MELLLA+ project. [[

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**References:**

RAI-08-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

RAI-08-2 NEDC-33075P-A, Revision 6, "LTR General Electric Boiling Water Reactor, Detect and Suppress Solution – Confirmation Density," January 2008.

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**Reactor Systems RAI-09**

*Section 2.4.3 of NEDC-33435P/Rev1 mentions two BSP options. Specify which BSP option will be implemented in Monticello. Provide a copy of the relevant sections in the Monticello Technical Specifications. Specifically, what is the maximum period of time that Monticello will be allowed under BSP conditions without the primary DSS-CD option operable?*

**GEH Response**:

Section 2.4.3 of NEDC-33435P, Revision 1 (Reference RAI-09-1) is consistent with Option 2 of NEDC-33075P-A, Revision 6 (Reference RAI-09-2), Section 7.5.2, which includes use of an Automated Backup Stability Protection (ABSP). The BSP Solution is implemented as documented in the proposed Monticello Technical Specifications (TS). Those proposed TS changes are consistent with the recommended TS changes provided in Reference RAI-09-2, and are provided in Attachment 1 of Xcel Energy Letter, L-MT-10-003 (Reference RAI-09-3).

[[

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**References:**

RAI-09-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

RAI-09-2 NEDC-33075P-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," January 2008.

RAI-09-3 Xcel Energy Letter, L-MT-10-003, "License Amendment Request: Maximum Extended Load Line Limit Analysis Plus," dated January 21, 2010

**Reactor Systems RAI-10**

*Section 2.4.1 and some tables and figures in NEDC-33435P/Rev1 reference TRACG002 results and criteria. Others contain references to TRACG004. Provide a summary of the code versions used for the Monticello FSAR analyses. Provide a short discussion of the licensing applicability of each code version, and specifically discuss the use of TRACG002 versus TRACG004. Section 2.6.1 states that the most recent versions of TGBLA/PANAC were used for the analyses. Specify which versions were used and discuss any interface issues with older codes like TRACG002.*

**GEH Response**:

A summary of the code versions (including TRACG versions) used for the Monticello FSAR analyses in Section 2.4.1 and associated tables and figures is provided in Table 1-1 of NEDC-33435P Revision 1 (Reference RAI-10-1). See the Thermal-Hydraulic Stability task row entries.

The Notes in Table 1-1 of Reference RAI-10-1 provide a short discussion of the licensing applicability of each code version.

The discussion about the use of TRACG02 versus TRACG04 was provided in a presentation shared with NRC staff reviewers on May 4, 2010. The NRC staff reviewers acknowledged that the content of the presentation provides sufficient explanation on the use of TRACG02 versus TRACG04 and the applied methodology. The NRC staff reviewers also mentioned that Monticello specific values and results should be used in the presentation material. Therefore, the provided presentation was updated to include Monticello MELLLA+/DSS-CD specific values and results as documented in Reference RAI-10-1. The resulting updated presentation from the May 4, 2010 NRC-GEH meeting is included on the Enclosure 3 CD ROM with the following file name:

"001_DSS-CD_TRACG02_Application_Method_to_MNGP.pdf"

Note that this file is also referenced in the responses to RAI-06 and RAI-08.

The TGBLA/PANAC versions that are used for developing the Monticello equilibrium core described in Section 2.6.1 is provided in Table 1-1 of Reference RAI-10-1. There are no interface issues with older codes like TRACG02 because the Monticello equilibrium core described in Section 2.6.1 was not used to run any TRACG02 analysis.

**Reference:**

RAI-10-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

## Reactor Systems RAI-11

*Table 9-1 of NEDC-33435P/Rev1 shows the AOO results in terms of peak power, flux, pressure and delta-CPR. The turbine trip with bypass (TTWBP) AOO appears to be the limiting delta-CPR event. The peak power during over-pressure events is typically very sensitive to the steam separator inertia (L/A) values used. Justify the steam separator L/A values used for these analyses.*

## GEH Response:

The ODYN computer code is used to perform the analysis for the overpressure events. The NRC approved report for ODYN (Reference RAI-11-1) includes details of the steam separator model used in the ODYN computer code. The steam separator L/A is a function of separator inlet quality and the relationship between separator inlet quality and L/A determined in qualification testing was input into the ODYN computer code.

## Reference:

RAI-11-1. Licensing Topical Report, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154-A, Vol. 1 and 2, August 1986 and NEDE-24154-P-A, Vol. 3, August 1988.

**Reactor Systems RAI-12**

*Table 9-1 of NEDC-33435P/Rev1 indicates that the turbine trip with bypass results in a higher peak power and lower CPR margin that generator load rejection or turbine trip without bypass. Provide an explanation why the bypass-failed transients result in a smaller power peak than the bypass available condition.*

**GEH Response**:

The Turbine Trip with Bypass (TTWBP) event results in a higher peak power and lower CPR margin than the generator load rejection or turbine trip without bypass because it was analyzed with a degraded scram speed associated with a postulated scenario involving a Turbine Trip with a reduced air volume in the scram discharge header. In the analysis, there is no credit taken for the faster scram speeds associated with Option B. Therefore, the Option A and Option B CPR results for the TTWBP transient are identical. The TTWBP is not limiting for Option A, but because no credit is taken for the faster scram speeds associated with Option B, the TTWBP becomes the limiting transient when compared to other transients that credit the faster scram times associated with the Option B OLMCPR. If the Option B scram time basis was used in the TTWBP, the Turbine Trip with no bypass would bound the results for the TTWBP. The Option A and Option B approaches are explained in Reference RAI-12-1, on pages US.C-196 and US.C-197.

**Reference:**

RAI-12-1. NEDE-24011-P-A-16-US, "Licensing Topical Report, General Electric Standard Application for Reactor Fuel (GESTAR II), Supplement for United States," Global Nuclear Fuel – Americas, October 2007.

**Reactor Systems RAI-13**

*Section 9.1.1 of NEDC-33435P/Rev1 states that "Results for all AOO pressurization transient events analyzed, including equipment out-of-service, showed at least 10% margin to the fuel centerline melt and the 1% cladding circumferential plastic strain acceptance criteria." Provide a table with the actual margins.*

**GEH Response**:

Margins for fuel centerline melt and 1% cladding circumferential strain were calculated for the EPU power level and reported in Reference RAI-13-1, Section 2.8.5. The most limiting flow condition for the pressurization transient events analyzed with respect to these criteria is increased core flow. As a result, the actual margins are unchanged from those provided in Reference RAI-13-1 because the MELLLA+ condition represents a less limiting, lower flow at the same EPU power level. The minimum calculated margin to the fuel centerline melt criterion was reported as 26%. The minimum calculated margin to the cladding strain criterion was reported as 35%. These reported values were derived from the results of the limiting pressurization transient with respect to the fuel centerline melt and cladding strain criteria, the inadvertent High Pressure Coolant Injection (HPCI) with L8 turbine trip at the increased core flow condition (105% of rated core flow).

**Reference:**

RAI-13-1. NEDC-33322P, Revision 3, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," October 2008.

**Reactor Systems RAI-14**

*Provide the results of the slow recirculation flow increase mentioned in Section 9.1.2 of NEDC-33435P/Rev1 and compare them with the MCPR flow factor.*

**GEH Response**:

Table RAI-14-1 summarizes the results of the slow recirculation flow increase analysis mentioned in Section 9.1.2 of NEDC-33435P, Revision 1 (Reference RAI-14-1) and compares them with the MCPR flow limit. The limit accounts for the ECCS-LOCA minimum initial MCPR for EPU/MELLLA+ operation of 1.35. The reference limits bound the slow recirculation flow results performed for the MELLLA+ operating domain.

**Reference:**

RAI-14-1 NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

**Table RAI-14-1: Comparison Slow Recirculation Flow Increase Results and MCPR Flow Limit**

| Flow (%) | Slow Recirculation Flow Increase MCPR | MCPR Flow Limit |
|---|---|---|
| 57.4 | 1.282 | 1.40 |
| 60.0 | 1.278 | 1.38 |
| 70.0 | 1.256 | 1.35 |
| 80.0 | 1.228 | 1.35 |
| 90.0 | 1.195 | 1.35 |
| 102.5 | 1.148 | 1.35 |
| 107.0 | 1.130 | 1.35 |

**Reactor Systems RAI-15**

*For the licensing ODYN ATWS analysis and the best estimate TRACG analysis described in Section 9.3 of NEDC-33435P/Rev1, provide time traces and tabulated values for reactor power, pressure, peak PCT, and suppression pool temperature. Provide the HCTL as function of reactor pressure and the HSBW injection time.*

**GEH Response**:

ODYN ATWS Analysis

The tabulated peak value and time trace for reactor power, reactor dome pressure, Peak Cladding Temperature (PCT) and suppression pool temperature is provided below for the limiting event in the ODYN ATWS analysis. For reactor power, analysis results are provided for the limiting event with respect to peak reactor vessel pressure. The limiting event is the Pressure Regulator Failure Open (PRFO) at Beginning Of Cycle (BOC) or End Of Cycle (EOC).

| **Parameter** | **Limiting Event** | **Peak Value** | **Time Trace** |
|---|---|---|---|
| Reactor Power (Neutron Flux) | PRFO at BOC | 289% Rated | Figure RAI-15-1 |
| Reactor Dome Pressure | PRFO at BOC | 1452 psia | Figure RAI-15-2 |
| Suppression Pool Temperature | PRFO at EOC | 197°F | Figure RAI-15-3 |
| Peak Cladding Temperature | PRFO at EOC | 1402°F | Figure RAI-15-4 |

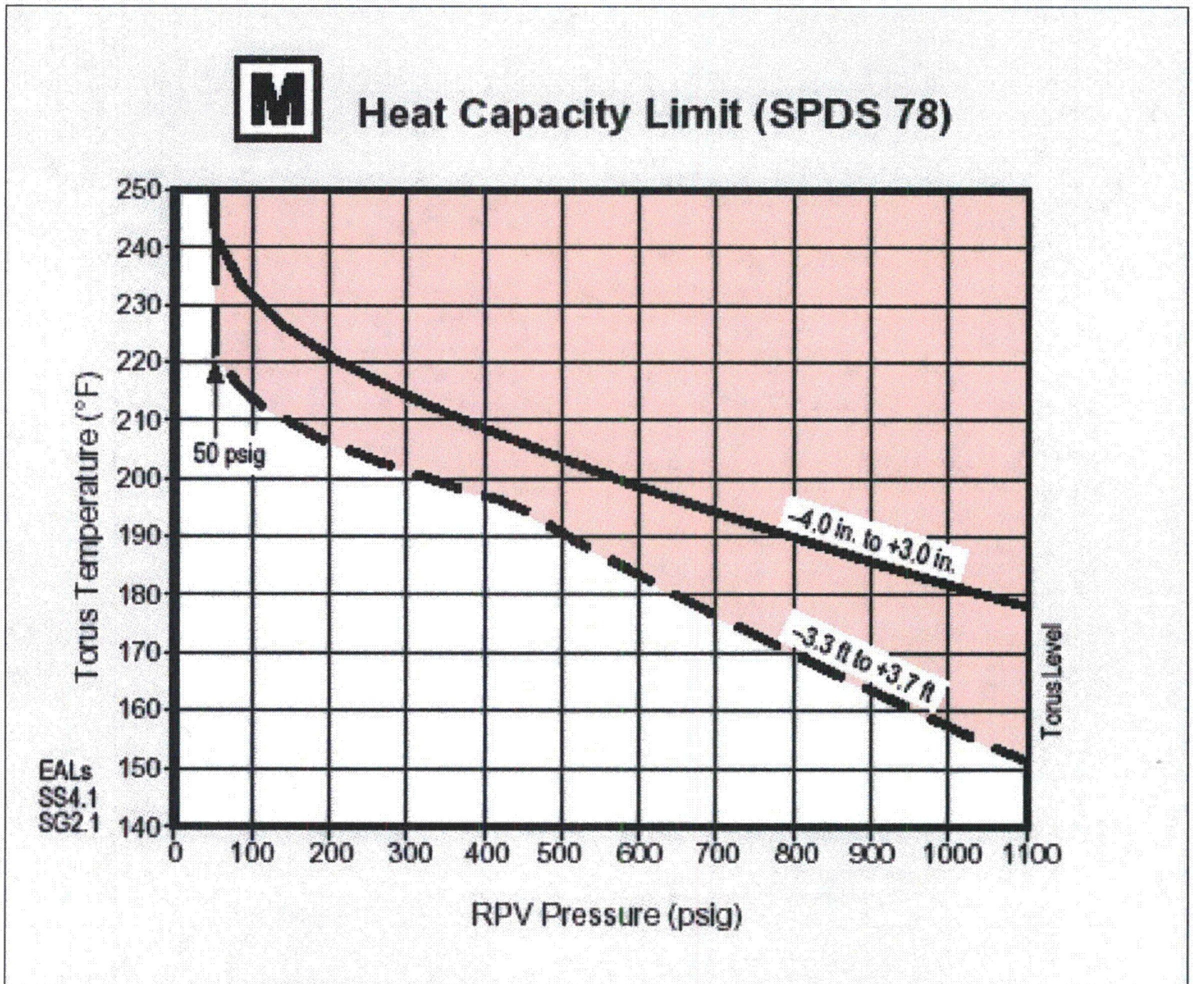
Best-Estimate TRACG ATWS Analysis

The tabulated peak value and time trace for reactor power, reactor dome pressure, PCT and suppression pool temperature is provided below for the best-estimate TRACG ATWS Main Steam Isolation Valve Closure (MSIVC) at EOC with a Heat Capacity Temperature Limit (HCTL) of 175°F and water level strategy at Top of Active Fuel (TAF). An HCTL of 175°F corresponds to the HCTL at a pressure near the Safety/Relief Valve (SRV) lift pressure and normal suppression pool water level.

| **Parameter** | **Limiting Event** | **Peak Value** | **Time Trace** |
|---|---|---|---|
| Reactor Power | MSIVC at EOC<br>HCTL of 175°F<br>Water Level<br>Strategy at TAF | 196% Rated | NEDC-33435P/Rev 1<br>Figure 9-7 |
| Reactor Dome Pressure | MSIVC at EOC<br>HCTL of 175°F<br>Water Level<br>Strategy at TAF | 1375 psia | NEDC-33435P/Rev 1<br>Figure 9-9 |
| Suppression Pool Temperature | MSIVC at EOC<br>HCTL of 175°F<br>Water Level<br>Strategy at TAF | 174°F | NEDC-33435P/Rev 1<br>Figure 9-10 |
| Peak Cladding Temperature | MSIVC at EOC<br>HCTL of 175°F<br>Water Level<br>Strategy at TAF | 719°F | NEDC-33435P/Rev 1<br>Figure 9-11 |

Heat Capacity Temperature Limit (HCTL)

The HCTL as a function of reactor pressure is provided below.




Reference: Monticello Nuclear Generating Plant EOP C.5-1200 Revision 16.

It is noted that the curve provided above changes by a small amount with EPU. However, the sensitivities on different HCTLs (175°F vs. 150°F) and water level strategies (TAF vs. TAF-2ft) were performed to evaluate the possible range of conditions at Monticello.

Hot Shutdown Boron Weight (HSBW) Injection Time

The HSBW injection time for Monticello is 19.2 minutes. This is the HSBW injection time for both MELLLA+ and non-MELLLA+ conditions.

Figure RAI-15-1
ODYN ATWS Analysis – PRFO at Beginning Of Cycle (BOC)
Reactor Power (Neutron Flux)

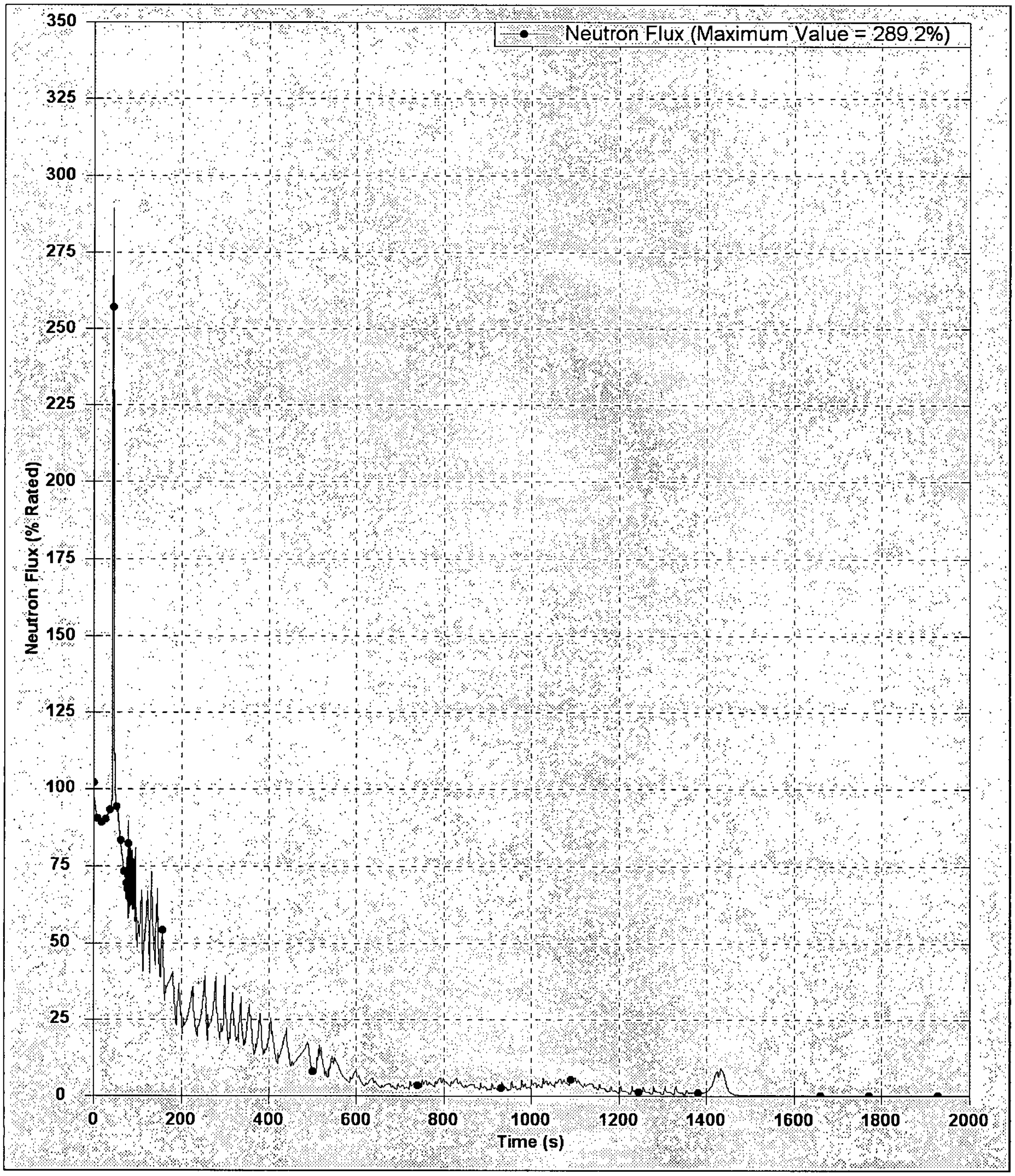


Figure RAI-15-2
ODYN ATWS Analysis – PRFO at Beginning Of Cycle (BOC)
Reactor Dome Pressure

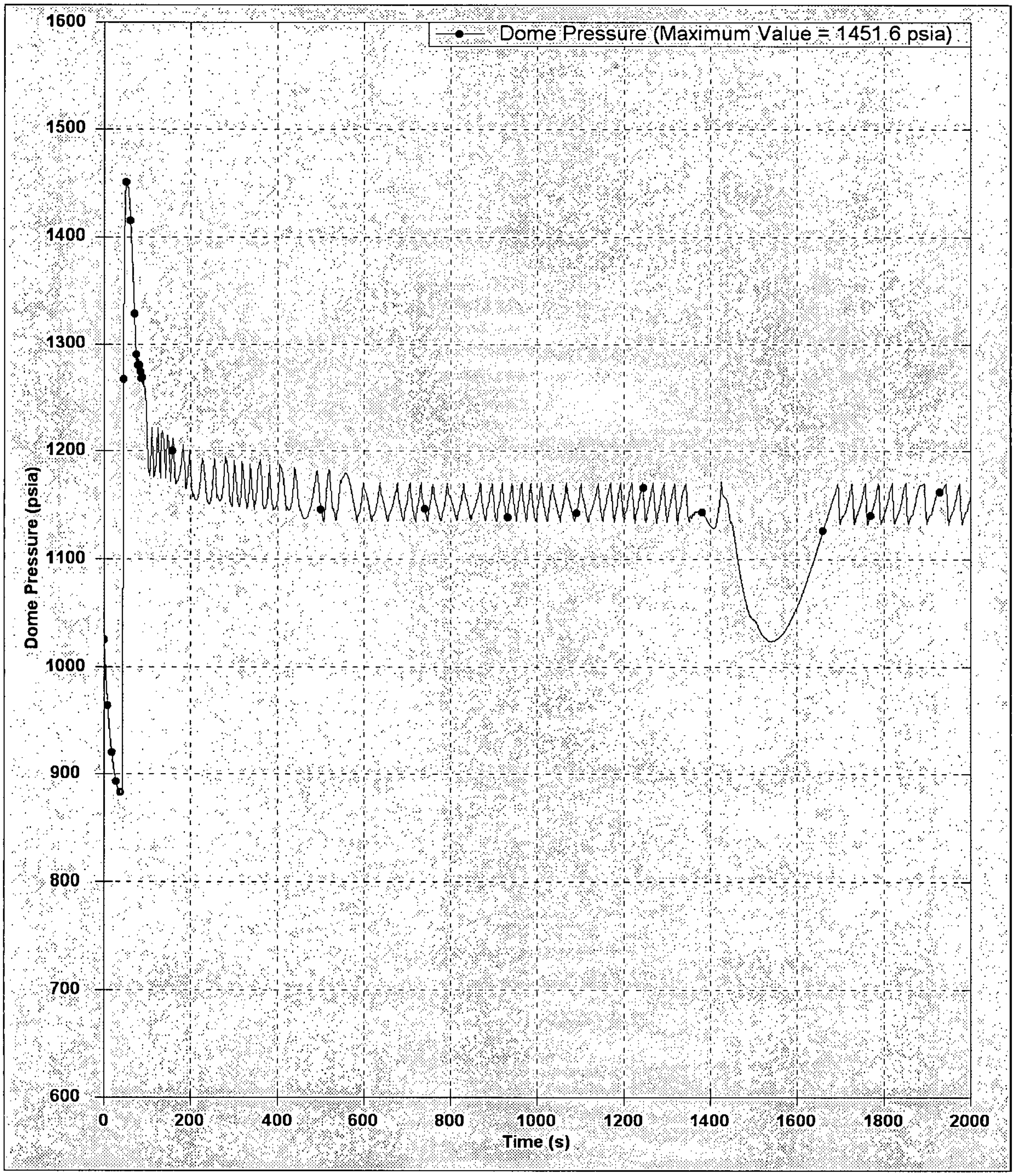


Figure RAI-15-3
ODYN ATWS Analysis – PRFO at End Of Cycle (EOC)
Suppression Pool Temperature

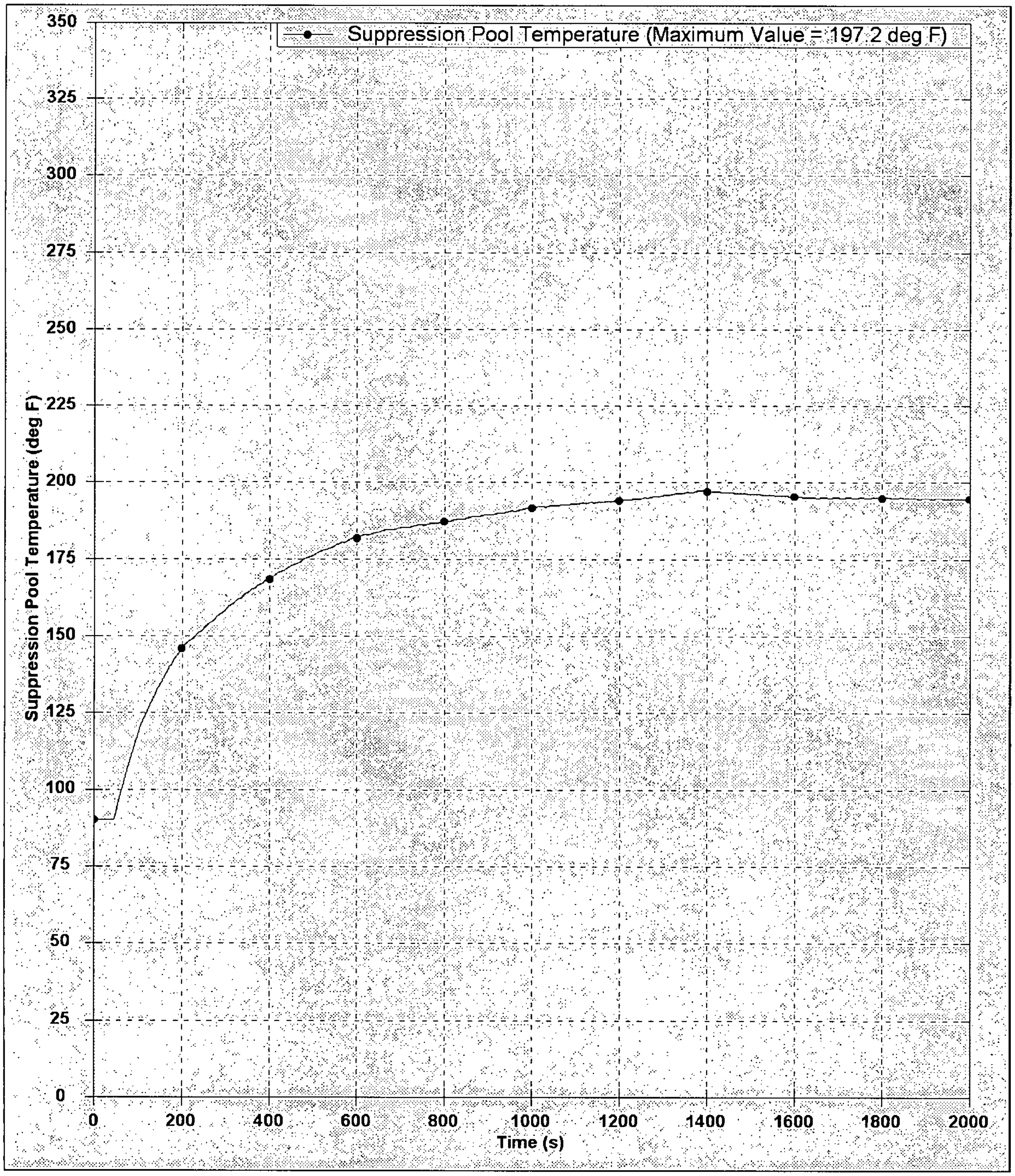
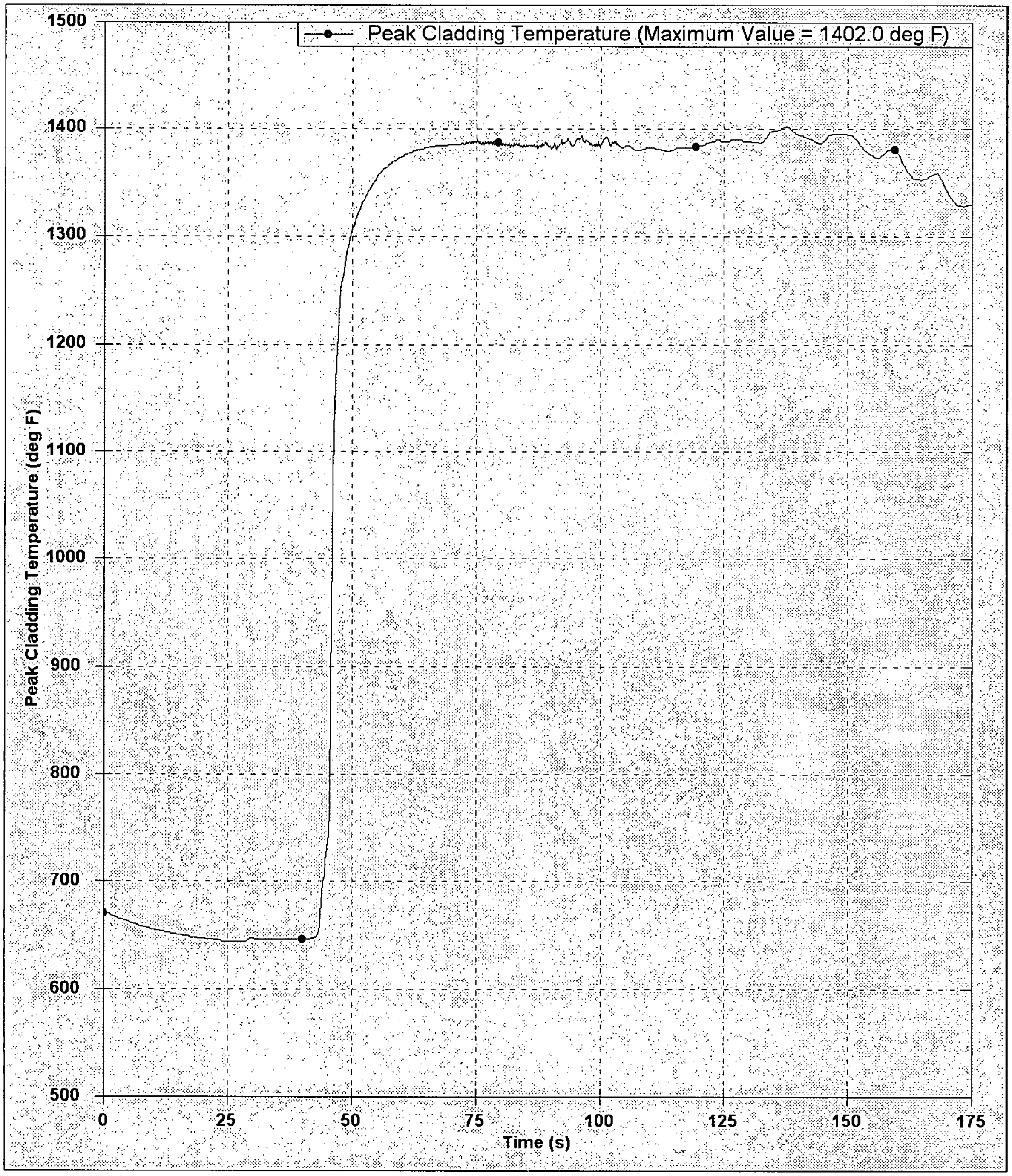


Figure RAI-15-4
ODYN ATWS Analysis – PRFO at End Of Cycle (EOC)
Peak Cladding Temperature



**Reactor Systems RAI-16**

*The Heat Capacity Temperature Limit (HCTL) is set to provide sufficient temperature margin in the suppression pool so that a conservative blow-out of the vessel with that initial suppression pool temperature will not result in a final temperature that compromises containment limits. Therefore, the HCTL limit is a function of operating reactor pressure. How is the HCTL limit determined in Monticello? Why are two arbitrary HCTL values of 150F and 175F used in the ATWS analyses of Section 9.3.1.2 of NEDC-33435P/Rev1? How do these values compare with the actual HCTL limit?*

**GEH Response**:

The HCTL is determined from the Monticello Emergency Operating Procedures (EOPs). The HCTL is a function of operating reactor pressure and suppression pool water level. For normal suppression pool water level, the HCTL is approximately 175°F near the SRV opening pressure. At the extreme upper or lower suppression pool water levels covered by EOPs, the HCTL is approximately 150°F at an opening pressure near the SRV opening pressure.

For the best-estimate ATWS TRACG analysis, the baseline event analyzed is the Main Steam Isolation Valve Closure (MSIVC) event with an HCTL of 175°F with a water level strategy that controls level at Top of Active Fuel (TAF). Sensitivities on different HCTLs (175°F vs. 150°F) and water level strategies (TAF vs. TAF-2ft) were performed to evaluate the possible range of operating conditions at Monticello.

**Reactor Systems RAI-20**

*Since the MELLLA+ SER was issued, a number of Part 21 notifications have been issued and evaluated. These issues are not part of the accepted SER, but have safety relevance to Monticello operation in the MELLLA+ domain. Provide a list of the applicable Part 21 issues that have been issued since the approval of the MELLLA+ SER and may affect MELLLA+ operation and a short description of their disposition.*

**GEH Response**:

Part 21 notifications issued since the MELLLA+ Safety Evaluation Report (SER) was issued were reviewed. (The MELLLA+ SER is contained in NEDC-33006P-A, Revision 3, Reference RAI-20-1). The Part 21 notifications reviewed are located on the NRC web site.

Part 21 notifications involving components were judged to be covered by utility Part 21 programs in that the component issue must be individually dispositioned to assure safe operation independent of MELLLA+.

Part 21 notifications that pertain to limitations in methodology were reviewed to determine whether they had been resolved for MELLLA+. Part 21s 2007-20-00 and 2007-20-01 involve the non-conservatism in the GESTR-M Thermal-Mechanical Methodology and applies to NEDC-33173P and therefore has an effect on MELLLA+. NRC resolutions for these two Part 21s are documented in Appendix F of the SER for NEDC-33173P (Reference RAI-20-3) and applies to the EPU/MELLLA+ domain. The new requirement in accordance with these two Part 21s is applied to MELLLA+.

**References:**

RAI-20-1 NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," June 2009.

RAI-20-2 NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains," February 2006.

RAI-20-3 Letter, H. Nieh, NRR, to R. Brown, GEH, "Final Safety Evaluation for General Electric (GE)-Hitachi Nuclear Energy Americas, LLC (GHNE) Licensing Topical Report (LTR) NEDC-33173P, 'Applicability of GE Methods to Expanded Operating Domains'," dated January 17, 2008.

**Reactor Systems RAI-21**

*The NRC staff intends to perform confirmatory calculations of Monticello stability with the LAPUR code. Provide the following Monticello design data to support these calculations. Refer to Fig 1-1 of NEDC-33435P Rev1. Point A is defined in the figure and point A' is at the intersection of the natural circulation line and the MELLLA+ rod line.*

1. *Provide the Inlet loss coefficients for the Monticello channels. Provide the "ODYSY" combined loss coefficient, not the "TRACG" separate coefficients, along with the reference flow area for the K-values provided.*
2. *Point A for the last non-MELLLA+ Monticello core using equilibrium FW temperature, provide*
   - *a. Thermal power*
   - *b. Fraction of power deposited in the fuel*
   - *c. Total core flow*
   - *d. Bypass flow*
   - *e. 3D steady-state power distribution in digital form (i.e., axial node power for each bundle)*
   - *f. Core-average void reactivity coefficient (special PANACEA edit)*
   - *g. First harmonic mode sub-criticality*
3. *Point A' for a representative Monticello MELLLA+ core using equilibrium FW temperature, provide same information as the above point*
4. *Provide same information for Points A and A' above, but setting FW temperature a near vessel-pressure saturation conditions. This condition will simulate lowering the water level below the FW sparger and pre-heating the FW with vessel steam as required by EOPs. This condition will bound the stability during an ATWS event because it will over-estimate the power and flow by keeping the water level high.*

**GEH Response**:

1. *Provide the Inlet loss coefficients for the Monticello channels. Provide the "ODYSY" combined loss coefficient, not the "TRACG" separate coefficients, along with the reference flow area for the K-values provided.*

   For a Monticello core with all GE14 fuel, which is the current configuration of Monticello and the configuration used in the Monticello MELLLA+ evaluations, the ODYSY inlet Loss Coefficients are:

   For Point A,
   [[

   ]]

   For Point A',
   [[

   ]]

2. *Point A for the last non-MELLLA+ Monticello core using equilibrium FW temperature, provide*

   *a. Thermal power*

   *b. Fraction of power deposited in the fuel*

   *c. Total core flow*

   *d. Bypass flow*

   *e. 3D steady-state power distribution in digital form (i.e., axial node power for each bundle)*

   *f. Core-average void reactivity coefficient (special PANACEA edit)*

   *g. First harmonic mode sub-criticality*

   Monticello Cycle 25 at EPU End of Rated exposure was analyzed at Non-MELLLA+ conditions. The parameters at point A for this core are

   [[

   ]]

3. *Point A' for a representative Monticello MELLLA+ core using equilibrium FW temperature, provide same information as the above point*

   The Monticello MELLLA+ core at End of Rated exposure at point A' has the following parameters

   [[

   ]]

4. *Provide same information for Points A and A' above, but setting FW temperature a near vessel-pressure saturation conditions. This condition will simulate lowering the water level below the FW sparger and pre-heating the FW with vessel steam as required by EOPs. This condition will bound the stability during an ATWS event because it will over-estimate the power and flow by keeping the water level high.*

   The cores described in Items 2 and 3 were analyzed by setting the inlet enthalpy to the saturation enthalpy based on the dome pressure and holding the reactor pressure constant. For Point A, the following parameters were calculated:

   [[

   ]]

   For Point A', the following parameters were calculated:

   [[

   ]]

**Reactor Systems RAI-22**

*Please explain why the PCT for the top peaked axial power distribution produces a lower PCT than the mid-peak distribution in the Table for section 4.3.2 corresponding to the 100 / 80 condition under the Appendix K column. Also, from this Table, please explain why the mid-peak from the 100 / 80 condition produces a lower PCT than that for the top-peak PCT at the 100 / 100 condition under the Appendix K column. Lastly, please explain why the top-peak 100 / 100 condition is higher than the top-peak at 100 / 80 condition under the Appendix K column. The lower flow rate would be expected to reduce the subcooled level in the core, increase the boiling length, decreasing the two-phase level and increasing PCT.*

**GEH Response**:

(1) *Please explain why the PCT for the top peaked axial power distribution produces a lower PCT than the mid-peak distribution in the Table for section 4.3.2 corresponding to the 100 / 80 condition under the Appendix K column*

[[

]]

*(2) Also, from this Table, please explain why the mid-peak from the 100 / 80 condition produces a lower PCT than that for the top-peak PCT at the 100 / 100 condition under the Appendix K column*

[[

]]

*(3) Lastly, please explain why the top-peak 100 / 100 condition is higher than the top-peak at 100 / 80 condition under the Appendix K column. The lower flow rate would be expected to reduce the subcooled level in the core, increase the boiling length, decreasing the two-phase level and increasing PCT.*

[[

]]

**Reactor Systems RAI-23**

*In section 4.3.8, please quantify what "small change" in PCT means. Also, it is stated that because the drive flow mismatch is small compared to MELLLA+, the PCT change due to the drive flow mismatch is expected to be smaller than the MELLLA+ sensitivity. What is this sensitivity? Was the PCT change demonstrated to be smaller through analyses? If so, what are the results? Please explain.*

**GEH Response**:

The flow mismatch issue and sensitivity is generically addressed in Reference RAI-23-1 and is addressed in the SER for Reference RAI-23-1, in Sections 4.3.7 and 4.3.8.

[[

]]

**Reference:**

RAI-23-1 NEDC-33006P-A, "Licensing Topical Report General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," Revision 3, June 2009.

**Reactor Systems RAI-24**

*Please provide the analysis results for the Appendix K results in the section 4.3.2 table for the top and mid peaked axial power distributions.*

**GEH Response**:

The mid-peaked hot channel axial power shape used in the Appendix K EPU rated (100P/100F) case is:

[[

]]

The top-peaked hot channel axial power shape used in the Appendix K EPU rated (100P/100F) case is:

[[

]]

The mid-peaked hot channel axial power shape used in the Appendix K MELLLA+ (100P/80F) case is:

[[

]]

The top-peaked hot channel axial power shape used in the Appendix K MELLLA+ (100P/80F) case is:

[[

]]

**Reactor Systems RAI-26**

*Please provide the following information relating to the Monticello MELLLA+ operation:*

(1) *details to obtain a final core loading pattern including procedure, guidance, criteria, and approved methodologies used for this analysis.*

(2) *when the final or reference core loading pattern will be available for analyzing the cycle-specific operating limits listed in the Table of Section 2.2.*

(3) *when the final reload analysis report will be available for parameters listed in Sections 2.3, 2.4, and 2.5 in the reload analysis report.*

**GEH Response**:

(1) *details to obtain a final core loading pattern including procedure, guidance, criteria, and approved methodologies used for this analysis.*

The core loading pattern is initially developed by Northern States Power – Minnesota (NSPM). The loading pattern is sent to Global Nuclear Fuel (GNF) for analysis in accordance with GESTAR II. GESTAR II is the umbrella for all procedures, guidelines, criteria, and approved methodologies used for GNF's analysis. GNF checks that the core loading pattern complies with the required inputs. Among the inputs are:

- Cycle energy
- Batch size
- Fuel bundle designs (nuclear)
- Core loading pattern
- Thermal limit margins
- Reactivity margins – minimum shutdown margin, minimum and maximum hot excess reactivity, standby liquid control system margin
- Discharge exposure limitations and other limits as established by safety analysis
- Desired control rod patterns – sequences and durations
- Channel bow acceptably minimized

If necessary, GNF recommends changes to the core design. GNF confirms that the final core design meets all of the criteria required by GESTAR II using GNF's approved methodologies.

During a conference call on July 19, 2010, the NRC staff requested the core design for the Monticello Cycle 26 core. The Cycle 26 core design is provided by NSPM in Attachment 1.

*(2) when the final or reference core loading pattern will be available for analyzing the cycle-specific operating limits listed in the Table of Section 2.2.*

The Cycle 26 reference core loading pattern is planned to be completed end of August 2010.

*(3) when the final reload analysis report will be available for parameters listed in Sections 2.3, 2.4, and 2.5 in the reload analysis report.*

The final reload analysis report is planned to be made available in May 2011. This is consistent with the NRC assessment of core reload requirements in the MELLLA+ SER, which includes a commitment to submit the SRLR for initial MELLLA+ implementation for NRC staff confirmation.

**Reactor Systems RAI-27**

*Please provide clarification for the relationship between footnote (b) in Table 3.3.1.1-1 and Function 2.b in term of the RTP and footnote (h) in Attachment 1 of L-MT-10-003.*

**GEH Response**:

The two notes are not related. Note b is related to single loop operation and note h is related to a stability issue.

As stated in Section 2.4.3 of NEDC-33435P, Revision 1 (Reference RAI-27-1) Monticello has chosen to utilize the Automated Backup Stability Protection (ABSP) Scram Region in the event that the primary stability protection afforded by the DSS-CD licensing basis algorithm (Confirmation Density Algorithm) is not operable (OPRM Function 2.f INOP). Section 7.5.2 of NEDC-33075P-A, Revision 6 (Reference RAI-27-2), describes the BSP option that includes the ABSP. Section 7.4 of Reference RAI-27-2 describes the ABSP.

To implement the ABSP, a change in the allowable value in Function 2.b, Average Power Monitors, Simulated Thermal Power – High, would be required. Footnote (h) was added to address such a change and is consistent with the recommended Technical Specifications provided in Reference RAI-27-2. Table 8-1 of Reference RAI-27-2 documents the basis for each recommended change to address the use of DSS-CD.

Note b also applies to Function 2.b but is not related to stability issues. That note addresses the change in the allowable value for Function 2.b due to single loop operation (SLO). Technical Specification 3.4.1 addresses SLO. The changes in the allowable values in SLO are required to maintain consistency with the assumptions of the Monticello SAFER/GESTR-LOCA Loss of Coolant Accident Analysis.

**References:**

RAI-27-1 GEH Report, NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," December 2009.

RAI-27-2 GEH Report, NEDC-33075P-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," January 2008.

ENCLOSURE 4

GE-MNGP-AEP-1913 R1

Affidavit

## GE-Hitachi Nuclear Energy Americas LLC

## AFFIDAVIT

**I, Edward D. Schrull,** state as follows:

(1) I am the Vice President, Regulatory Affairs, Services Licensing, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH letter, GE-MNGP-AEP-1913 R1, B. Hagemeier, GEH, to A. Williams, Northern States Power - Minnesota, "GEH Responses to NRC Reactor Systems RAIs," dated August 27, 2010. The proprietary information in Enclosure 1 entitled, "GEH Responses to Reactor Systems RAIs – Proprietary," is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]]. The entirety of the information in Enclosure 3 entitled, "Compact Disc (CD) – Proprietary," is considered proprietary; in particular, the contents of the following files are considered proprietary:

- 001_DSS-CD_TRACG02_Application_Method_to_MNGP.pdf
- 002_Monticello_3D_Power_Point_A.xls
- 003_Monticello_3D_Power_Point_AP.xls
- 004_Monticello_3D_Power_Point_A_Sat_FW.xls
- 005_Monticello_3D_Power_Point_AP_Sat_FW.xls

The disk itself is marked as "GEH Proprietary Information." In all cases, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission</u>, 975 F2d 871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA</u>, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains results of an analysis performed by GEH to support Monticello Nuclear Generating Plant Maximum Extended Load Line Limit Analysis Plus (MELLLA+) license application. This analysis is part of the GEH MELLLA+ methodology. Development of the MELLLA+ methodology and the supporting analysis techniques and information, and their application to the design, modification, and processes were achieved at a significant cost to GEH.

The development of the evaluation methodology along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 27th day of August 2010.

Edward D. Schrull
Vice President, Regulatory Affairs
Services Licensing
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# ATTACHMENT 3

Markup and Final Replacement Page for Enclosure 1, Table 1 (page 4 of 14) of Reference 1, "License Amendment Request: Maximum Extended Load Line Limit Analysis Plus," L-MT-10-003, dated January 21, 2010, TAC ME3145, ADAMS Accession No. ML100280558.

**2 pages follow**

# Enclosure 1

Markup showing correction

**Table 1**
**Monticello Proposed**
**Operating License and Technical**
**Specification Changes**

| TS Section | Description of Change | Basis for Change |
|---|---|---|
| TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation<br>ACTIONs I, J, and K | Modify ACTIONs I, J, and K and to reflect new Backup Stability Protection (BSP) requirements. | Attachment 3, Section 2.4.3 and as described in the DSS-CD LTR (Reference 2) |
| TS 3.3.1.1, "Oscillation Power Range Monitor (OPRM) Instrumentation" Surveillance Requirements<br>SR ~~3.1.1.1.16~~ 3.3.1.1.16 | (Delete the Surveillance.) | Deleted to eliminate unnecessary actions.<br>DSS-CD LTR (Reference 2). |
| Table 3.3.1.1-1, APRM Function 2b<br>Simulated Thermal Power - High<br>ALLOWABLE VALUE | Revises the Allowable Value to ≤ 0.61W + 67.2% RTP.<br>Add new note (h) to require resetting the APRM-STP – High when the OPRM (Function 2f) is INOP. | Attachment 3, Section 5.3.1<br>Attachment 3, Section 2.4.3 |
| Table 3.3.1.1-1, APRM Function 2f<br>OPRM Upscale | Add note describing initial implementation arming requirements. | DSS-CD LTR (Reference 2), Section 2.1 |
| TS 3.4.1, "Recirculation Loops Operating"<br>LCO 3.4.1 | Restrict MELLLA+ Operation in Single Recirculation Loop configuration. | MELLLA+ is not analyzed for Single Loop Operation<br>Attachment 3, Section 3.6.3 |
| TS 5.6.3 Core Operating Limits Report (COLR)<br>a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: | Delete Item 6, the requirement for the Period Based Detection Algorithm Setpoints and replace with:<br>6. The Manual Backup Stability Protection (BSP) Scram Region (Region I), The Manual BSP Controlled Entry Region (Region II), the modified APRM-STP setpoints used in the Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1. | Adding DSS-CD LTR requirements (Reference 2) |
| b The analytical methods used to determine the core operating limits shall be those | Replace Item 4, with: NEDO- | |

# Enclosure 1

| Table 1<br>Monticello Proposed<br>Operating License and Technical<br>Specification Changes | | |
|---|---|---|
| **TS Section** | **Description of Change** | **Basis for Change** |
| TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation<br>ACTIONs I, J, and K | Modify ACTIONs I, J, and K and to reflect new Backup Stability Protection (BSP) requirements. | Attachment 3, Section 2.4.3 and as described in the DSS-CD LTR (Reference 2) |
| TS 3.3.1.1, "Oscillation Power Range Monitor (OPRM) Instrumentation" Surveillance Requirements<br>SR 3.3.1.1.16 | (Delete the Surveillance.) | Deleted to eliminate unnecessary actions.<br>DSS-CD LTR (Reference 2). |
| Table 3.3.1.1-1, APRM Function 2b<br>Simulated Thermal Power - High<br>ALLOWABLE VALUE | Revises the Allowable Value to ≤ 0.61W + 67.2% RTP.<br>Add new note (h) to require resetting the APRM-STP – High when the OPRM (Function 2f) is INOP. | Attachment 3, Section 5.3.1<br>Attachment 3, Section 2.4.3 |
| Table 3.3.1.1-1, APRM Function 2f<br>OPRM Upscale | Add note describing initial implementation arming requirements. | DSS-CD LTR (Reference 2), Section 2.1 |
| TS 3.4.1, "Recirculation Loops Operating"<br>LCO 3.4.1 | Restrict MELLLA+ Operation in Single Recirculation Loop configuration. | MELLLA+ is not analyzed for Single Loop Operation<br>Attachment 3, Section 3.6.3 |
| TS 5.6.3 Core Operating Limits Report (COLR)<br>a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:<br>b The analytical methods used to determine the core operating limits shall be those | Delete Item 6, the requirement for the Period Based Detection Algorithm Setpoints and replace with:<br>6. The Manual Backup Stability Protection (BSP) Scram Region (Region I), The Manual BSP Controlled Entry Region (Region II), the modified APRM-STP setpoints used in the Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.<br>Replace Item 4, with: NEDO- | Adding DSS-CD LTR requirements (Reference 2) |