Exelon Generation.

Attachments 5, 8, 10 and 14 transmitted herewith contain ~~Proprietary Information~~. Attachment 13 transmitted herewith contains Critical Energy Infrastructure Information (CEII). When separated from Attachments 5, 8, 10, 13 and 14, this document is decontrolled.

10 CFR 50.90
10 CFR 50, Appendix K
10 CFR 2.390

February 17, 2017

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

Peach Bottom Atomic Power Stations, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Request for License Amendment Regarding Measurement Uncertainty Recapture Power Uprate

Reference: NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of 65 megawatts thermal (MWt) (approximately 1.66%) in rated thermal power (RTP) from the current licensed thermal power (CLTP) of 3951 MWt to 4016 MWt.

The proposed changes are based on taking credit for the increased feedwater flow measurement accuracy achieved with the Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. Although installed at Unit 2 and Unit 3 in 2002 and 2003, respectively, in implementation of the Measurement Uncertainty Recapture (MUR) license amendments received in 2002, the current licensing basis analyses have not taken credit for the improved accuracy of the LEFM since receipt of the Extended Power Uprate (EPU) license amendments in 2014.

Attachments 5, 8, 10 and 14 transmitted herewith contain ~~Proprietary Information~~. Attachment 13 transmitted herewith contains Critical Energy Infrastructure Information (CEII). When separated from Attachments 5, 8, 10, 13 and 14, this document is decontrolled.

The content of this request is consistent with the guidance contained in the referenced RIS.

The proposed changes have been reviewed by the PBAPS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed changes by February 17, 2018. The requested review period is consistent with NRC internal guidance and supports business plan initiatives to increase EGC's generation capacity. Once approved, the amendment will be implemented within 90 days for both units. This implementation period will provide adequate time for revision of the affected station documents using the appropriate change control mechanisms and coordination with plant operating and maintenance activities.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the Commonwealth of Pennsylvania and the State of Maryland of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Officials.

Attachment 1 contains the evaluation of the proposed changes. Attachment 2 contains markups of the proposed Operating License and Technical Specifications pages. Attachment 3 contains markups of the proposed Technical Specifications Bases and Technical Requirements Manual pages (for information only). Attachment 4 contains the Regulatory Issue Summary 2002-03 cross-reference.

In accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," EGC requests withholding of: Attachments 5 and 14 (GE Hitachi Nuclear Energy Safety Analysis Report for PBAPS 2 and 3); Attachment 8 (Cameron Uncertainty Analyses for PBAPS 2 and 3); and Attachment 10 (Westinghouse Steam Dryer Report at MUR Conditions). GE Hitachi Nuclear Energy (GEH) and the Electric Power Research Institute (EPRI) request that Attachments 5 and 14 be withheld from public disclosure. A non-proprietary version of Attachment 5 is provided in Attachment 7. A non-proprietary version of Attachment 14 is not provided since all the non-proprietary information pertaining to this PBAPS submittal in Attachment 14 is provided in Attachment 7. Cameron is requesting that Attachment 8 be withheld, in its entirety, from public disclosure and thus a non-proprietary version is not provided. Westinghouse Electric Company (WEC) is requesting the Attachment 10 be withheld from public disclosure. A non-proprietary version of Attachment 10 is provided in Attachment 12. Affidavits in support of these requests are provided as Attachments 6 (GEH and EPRI), 9 (Cameron) and 11 (WEC).

Attachment 13 provides the grid stability study performed by the independent system operator at the expected full MUR electrical output and the voltage analysis by PECO Transmission Planning. These documents demonstrate that operation at the RTP requested herein will not have a significant effect on the reliability or operating characteristics of PBAPS or the offsite system. The contents of Attachment 13 contain critical energy infrastructure information (CEII) and are considered sensitive, unclassified (non-safeguard) information in accordance with 10 CFR 2.390(d)(1). As such, EGC requests that the information contained in Attachment 13 be withheld from public disclosure.

Attachment 14 provides a redline/strikeout version of the current GEH TSAR template that shows the changes made to generate the PBAPS-specific TSAR provided in Attachment 5. EGC also requests withholding of this attachment in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." An affidavit in support of this request is provided in Attachment 6.

Should you have any questions concerning this request, please contact Mr. David Neff at (610) 765-5631.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 17th day of February 2017.

Respectfully,

James Barstow

James Barstow
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Evaluation of Proposed Changes
2. Markup of Proposed Operating License and Technical Specifications Pages
3. Markup of Proposed Technical Specifications Bases and Technical Requirements Manual Pages (For Information Only)
4. NRC Regulatory Issue Summary 2002-03 Cross-Reference
5. GE Hitachi Nuclear Energy Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3 Thermal Power Optimization, NEDC-33873 (Proprietary Version)
6. GE Hitachi Nuclear Energy and EPRI Affidavits Supporting Withholding Attachments 5 and 14 from public disclosure
7. GE Hitachi Nuclear Energy Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3 Thermal Power Optimization, NEDO-33873 (Non-Proprietary Version)
8. Cameron ER-464, "Bounding Uncertainty Analysis for Thermal Power Determination at PBAPS Unit 2 Using the LEFM CheckPlus System," (Proprietary Version), and ER-463, "Bounding Uncertainty Analysis for Thermal Power Determination at PBAPS Unit 3 Using the LEFM CheckPlus System," (Proprietary Version)
9. Cameron Affidavit Supporting Withholding Attachment 8 from Public Disclosure
10. Westinghouse Electric Company, Peach Bottom Units 2 and 3 Steam Dryer Report at MUR Conditions, LTR-BWR-ENG-16-032-P, Revision 0 (Proprietary Version)
11. WEC Affidavit Supporting Withholding Attachment 10 from Public Disclosure
12. Westinghouse Electric Company, Peach Bottom Units 2 and 3 Steam Dryer Report at MUR Conditions, LTR-BWR-ENG-16-032-NP, Revision 0 (Non-Proprietary Version)
13. PJM Interconnection document, "Generator Transient Stability Study for Peach Bottom Atomic Power Station," and PECO document, Power Grid Voltage Analysis - Power Uprate Scenario for Peach Bottom Atomic Power Station."
14. Redline/Strikeout version of Attachment 5 (Proprietary)

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, PBAPS
USNRC Project Manager, PBAPS
R. R. Janati, Pennsylvania Bureau of Radiation Protection
S. T. Gray, State of Maryland

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Model," Exelon Generation Company, LLC (EGC), requests an amendment to Renewed Facility Operating License (RFOL) Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of 65 megawatts thermal (MWt) (approximately 1.66%) in rated thermal power (RTP) from 3951 MWt to 4016 MWt representing an overall increase of 22% from the original licensed thermal power level (OLTP) of 3293 MWt.

The generic applicability of the NRC-approved GE Thermal Power Optimization License Topical Report (NEDC-32938P-A) (Reference 1), commonly called the TLTR, is limited to a maximum Thermal Power Optimization (TPO) RTP of 120% of OLTP. It mandates that plants applying for a TPO uprate that would result in a licensed thermal power (LTP) in excess of 120% of OLTP provide plant-specific evaluations for those evaluations not performed at 102% of current licensed thermal power (CLTP). As described in Section 3.1 below, PBAPS Units 2 and 3 were originally licensed at 3293 MWt and received power uprates via amendments to the facility operating licenses as well as a Maximum Extended Load Line Limit Analysis Plus (MELLLA+) license amendment. As a result, the CLTP of the PBAPS units is now at 120% of OLTP.

Where required, the current safety analysis basis assumes that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level. The analyses performed at 102% of CLTP (i.e., 4030 MWt) remain applicable at the TPO RTP because the 2% factor from Regulatory Guide (RG) 1.49, “Power Levels of Nuclear Power Plants” (Reference 18), bounds the improvement in the FW flow measurements. Some analyses are performed at TPO RTP because the uncertainty factor is accounted for in the method or the additional 2% margin is neither required nor appropriate (e.g., reactor heat balance). In addition, some analyses (e.g., special events) are conservatively performed at the TPO bounding thermal power of 101.7% of CLTP (i.e., 4018 MWt). EGC is therefore providing plant-specific evaluations in support of this Measurement Uncertainty Recapture (MUR) LAR for those evaluations not performed at 102% of CLTP. These plant-specific evaluations use the current licensing basis which includes the amendments for Extended Power Uprate (EPU) and MELLLA+. Detailed descriptions of the basis for the TPO analyses are provided in Attachment 5.

The proposed changes are based on the increased feedwater flow measurement accuracy of the Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation which provides a more accurate calculation of reactor thermal power. The CheckPlus LEFM system was installed in the PBAPS Units 2 and 3 in 2002 and 2003, respectively, to support the first MUR power uprate license amendment request, which the NRC approved in November 2002. However, the analyses conducted in support of the EPU, approved in August 2014, assumed a 2% RTP uncertainty and did not take credit for the increased accuracy provided by the LEFM system.

2.0 DETAILED DESCRIPTION

The proposed changes to the Operating Licenses and Technical Specifications (TS) for Units 2 and 3 resulting from this MUR uprate are described in Sections 2.1 and 2.2 below. Marked-up pages are included in Attachment 2.

Proposed changes to the Technical Requirements Manual (TRM) resulting from this MUR uprate are described in Section 2.3. Marked-up pages of the TRM and TS Bases are provided in Attachment 3 for information only.

All changes to the TS and TS Bases required to implement TPO are provided in Attachments 2 and 3. Unless specifically addressed in these Attachments, no values of TSs or TS Bases that are based on the percent of RTP are changed.

2.1 Changes to the Operating Licenses

PBAPS, Units 2 and 3, RFOL Nos. DPR-44 and DPR-56, Sections 2.C(1), "Maximum Power Level," are revised to increase the value of RTP from 3951 MWt to 4016 MWt.

2.2 Changes to the Technical Specifications

Item	TS	TS Title	Description of Change
1.	1.1	Definitions	Change the value of RTP from “3951 MWt” to “4016 MWt”
2.	2.1.1.1	Reactor Core SLs	Re-scale the THERMAL POWER limit with the reactor dome pressure <700 psia or core flow < 10% rated core flow from 23% to 22.6% RTP to reflect the change as a result of the TPO.
3.	3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	Re-scale APPLICABILITY of LCO 3.2.1, REQUIRED ACTION B.1, and FREQUENCY for SR 3.2.1.1 from 23% to 22.6% RTP to reflect the change as a result of the TPO.
4.	3.2.2	MINIMUM CRITICAL POWER RATIO (MCPR)	Re-scale APPLICABILITY of LCO 3.2.2, REQUIRED ACTION B.1, and FREQUENCY for SR 3.2.2.1 from 23% to 22.6% RTP to reflect the change as a result of the TPO.
5.	3.2.3	LINEAR HEAT GENERATION RATE (LHGR)	Re-scale APPLICABILITY of LCO 3.2.3, REQUIRED ACTION B.1, and FREQUENCY for SR 3.2.3.1 from 23% to 22.6% RTP to reflect the change as a result of the TPO.
6.	3.3.1.1	Reactor Protection System (RPS) Instrumentation	In REQUIRED ACTION E.1, re-scale the turbine power/scram bypass setpoint from 26.7% to 26.3% RTP to reflect TPO.

Item	TS	TS Title	Description of Change
7.	3.3.1.1	Reactor Protection System (RPS) Instrumentation	In REQUIRED ACTION K.1, reduce the Oscillation Power Range Monitor (OPRM) Upscale Operable threshold from 18% to 17.6% RTP to reflect the TPO.
8.	3.3.1.1	Reactor Protection System (RPS) Instrumentation	In the NOTE for SR 3.3.1.1.2, re-scale the thermal limit monitoring threshold from 23% to 22.6% RTP in both places to reflect the TPO.
9.	3.3.1.1	Reactor Protection System (RPS) Instrumentation	In SR 3.3.1.1.13, re-scale the turbine power/scram bypass setpoint from 26.7% to 26.3% RTP to reflect the TPO.
10.	Table 3.3.1.1-1	Reactor Protection System Instrumentation	In Item 2b, change Allowable Value (AV) for Two Loop (TLO) Operation Average Power Range Monitor (APRM) Simulated Thermal Power (STP)-High due to TPO as follows: “0.61W + 67.1% RTP” to “0.60W + 65.9% RTP.”
11.	Table 3.3.1.1-1	Reactor Protection System Instrumentation	In Item 2f, reduce the OPRM Upscale Operable threshold from 18% to 17.6% RTP as a result of the TPO.
12.	Table 3.3.1.1-1	Reactor Protection System Instrumentation	In Note (b), change the AV for Single Loop Operation (SLO) APRM STP-High due to the TPO as follows: “0.55 (W – ΔW) + 61.5% RTP” to “0.54 (W – ΔW) + 60.3% RTP.”
13.	Table 3.3.1.1-1	Reactor Protection System Instrumentation	In Items 8 and 9, rescale the Applicable Modes or Other Specified Conditions column value from 26.7% to 26.3% RTP as a result of the TPO.
14.	3.3.2.2	Feedwater and Main Turbine High Water Level Trip Instrumentation	Re-scale APPLICABILITY and REQUIRED ACTION C.2 from 23% to 22.6% RTP to reflect the TPO.
15.	3.3.4.2	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	Re-scale APPLICABILITY, REQUIRED ACTION C.2, and Surveillance Requirement (SR) 3.3.4.2.4 for turbine power/scram bypass setpoint from 26.7% to 26.3% RTP as a result of the TPO.
16.	3.4.2	Jet Pumps	Re-scale Note 2 in SR 3.4.2.1 from 23% to 22.6% RTP as a result of the TPO.
17.	3.5.1.8	High Pressure Coolant Injection (HPCI) System	In SR 3.5.1.8, change the range of reactor pressures for verifying the HPCI pump’s ability to achieve the required flow rate from ≤ 1053 and ≥ 915 psig to ≤ 1053 and ≥ 910 psig.

Item	TS	TS Title	Description of Change
18.	3.5.3.3	Reactor Core Isolation Cooling (RCIC) System	In SR 3.5.3.3, change the range of reactor pressures for verifying the RCIC pump's ability to achieve the required flow rate from ≤ 1053 and ≥ 915 psig to ≤ 1053 and ≥ 910 psig.
19.	3.7.6	Main Turbine Bypass	Re-scale APPLICABILITY for LCO 3.7.6 and REQUIRED ACTION B.1 from 23% to 22.6% RTP to as a result of the TPO.

2.3 Changes to the Technical Requirements Manual

TRM Section	Title	Description of Change
1.1	Definitions	Revise the defined value of the RTP from 3951 MWt to 4016 MWt to reflect the TPO
Table 3.2-1	Control Rod Block Instrumentation	In Item 3, change the AV for SLO APRM STP-High due to the TPO for both TLO and SLO. The new equation for TLO is: $0.60\ W_d + 56.5\%$ The new equation for SLO is: $0.54\ (W_d - \Delta W) + 50.9\%$
3.6	Post-Accident Monitoring Instrumentation	In Test Requirement 3.6.2, re-scale the MCPR monitoring threshold for TPO power from 23% to 22.6% in two places
3.20 (New)	Leading Edge Flow Meter (LEFM) System	See Section 3.3.4 Criterion 1, *LEFM Inoperability* for a description

3.0 TECHNICAL EVALUATION

3.1. Background

PBAPS Units 2 and 3 have received the following license amendments authorizing increases in LTP:

- In 1994 and 1995, Amendments 198 and 211 to the Units 2 and 3 operating licenses, respectively, authorized a stretch power uprate of 5% from OLTP of 3293 MWt to 3458 MWt. (References 2 and 3).
- In 2002, Amendments 247 and 250 to the Units 2 and 3 operating licenses, respectively, authorized an MUR uprate from 3458 MWt to 3514 MWt based on the

reduced uncertainty in feedwater flow measurement using the installed LEFM systems (Reference 4).

- In 2014, Amendments 293 and 296 to the Units 2 and 3 operating licenses authorized an EPU increasing power from 3514 MWt to 3951 MWt (Reference 5).
- In 2016, Amendments 305 and 309 to the operating license authorized PBAPS to operate in the expanded MELLLA+ operating domain and to use the Detect and Suppress Solution – Confirmation Density (DSS-CD) stability solution (Reference 6).

3.2 General Approach

10 CFR 50, Appendix K, paragraph I.A, "Sources of heat during the LOCA," requires that emergency core cooling system (ECCS) evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 1, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

With credit for the LEFM system, the core thermal power measurement uncertainty will be a maximum of 0.34% and support an increase in RTP of 1.66% (i.e., 2.00% - 0.34%) from 3951 MWt to 4016.6 MWt which is conservatively rounded down to the requested 4016 MWt. Since this would result in an RTP level greater than 120% of OLTP, plant specific evaluations have been performed for those evaluations not previously conducted at 102% of CLTP.

EGC has evaluated the effects of a 65 MWt increase in RTP using an approach developed by GE Hitachi Nuclear Energy (GEH) and approved by the NRC, as documented in the TLTR (Reference 1). These evaluations are summarized in Section 3.5.1 below and described in detail in Attachment 5.

The scope and content of the evaluations performed and described in this request are consistent with the guidance contained in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," (Reference 7). Attachment 4 provides a cross-reference between the contents of this application and the guidance in RIS 2002-03.

3.3 LEFM Flow Measurement and Core Thermal Power Uncertainty

3.3.1 LEFM Flow Measurement

The LEFM system uses ultrasonic transit time principles to determine fluid velocity. This flow measurement method is described in topical reports ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System,” (Reference 8) and ER-157P, “Supplement to Topical Report ER-80P: Basis for Power Uprates with an LEFM √™ or LEFM CheckPlus ™ System, Revision 8,” (Reference 9).

In References 10 and 11, the NRC established criteria for use of these topical reports in requests for license amendments. EGC's response to those criteria is provided in Section 3.3.4, "Disposition of NRC Criteria for Use of LEFM Topical Reports."

Although the LEFM system is not safety-related, it is designed and manufactured in accordance with Cameron's Quality Assurance Program, which conforms to 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Cameron's verification and validation (V&V) program fulfills the requirements of ANSI/IEEE-ANS Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and ASME NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software V&V in EPRI TR-103291, "Handbook for Verification and Validation of Digital Systems," December 1994. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM system are provided in Reference 8, Section 6.4 and Table 6.1.

3.3.2 Plant Implementation

As indicated above, the NRC authorized an MUR power uprate from 3458 MWt to 3514 MWt in 2002, based on the use of the LEFM system to provide more accurate measurement of RTP. Actual measurement of RTP using the LEFM system began in 2002 and 2003 for Units 2 and 3, respectively, after startup from the refueling outages during which the LEFM systems were installed. In September 2014, the NRC authorized an EPU license amendment for PBAPS that increased the authorized licensed thermal power from 3514 MWt to 3951 MWt. Although the EPU analyses did not take credit for the reduced uncertainty provided by the LEFM system, the LEFM system remained the primary system to measure feedwater flow and provide input to the Core Thermal Power calculation. Plant procedures continued to ensure that the LEFM system was properly maintained, calibrated and operated. No changes have been made that would invalidate the calibration factors for the PBAPS spool pieces established by tests at Alden Research Laboratory in May 2002 for the original installation and MUR license amendment (Reference 4). The installed configuration at PBAPS on which this current application for an MUR uprate is based remains bounded by the original Caldon LEFM system installation and calibration assumptions as analyzed in the Caldon Topical Reports (References 8 and 9).

The LEFM system transducers are installed upstream of the original feedwater flow meters and are physically located outside of the 3rd and 4th stage feedwater heater rooms on elevation 135' in a designated locked high radiation area at power. The electronics cabinet is installed in the main lube oil equipment area in the turbine building on elevation 135', where the radiation field is generally <2 mR/hr at full power.

3.3.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology

Applying the results of testing and calibration of the LEFM system at PBAPS Units 2 and 3, and the methodology described in References 12 and 13, Cameron has calculated a feedwater mass flow rate uncertainty of ±0.30% with a fully functional LEFM system. With a feedwater mass flow uncertainty of ±0.30%, the total thermal power uncertainty is ±0.34%. The calculations for the feedwater and total system uncertainties for each unit are provided in Attachment 8. The calculation methodology is consistent with the PBAPS setpoint calculation methodology. The uncertainty is at a 95% probability and 95% confidence level.

3.3.4 Disposition of NRC Criteria for Use of LEFM Topical Reports

In References 10 and 11, the NRC established nine criteria to be addressed by licensees incorporating the LEFM methodology into the licensing basis. The nine criteria are listed below, along with a discussion of how each will be satisfied.

<u>Criterion 1</u>

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

<u>Response to Criterion 1</u>

<u>*Calibration and Maintenance*</u>

The necessary procedures and documents required for maintenance and calibration of the LEFM system have been implemented to ensure that the system is properly maintained and calibrated.

Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers are replaced as determined to be necessary by a review of the equipment's performance history by the LEFM system vendor.

For instrumentation other than the LEFM system that contributes to the power calorimetric computation, calibration and maintenance is performed periodically using existing site procedures. Maintenance and test equipment, setting tolerances, calibration frequencies, and instrumentation accuracy were evaluated and accounted for within the thermal power uncertainty calculations (Attachment 8).

<u>*LEFM Inoperability*</u>

The redundancy inherent in the two measurement planes of an LEFM system makes the system tolerant to component failures. The system features automatic self-testing. A continuously operating on-line test is provided to verify that the digital circuits are operating correctly and within the specified accuracy range.

A process exists to use the feedwater mass flow and temperature values from the corresponding LEFM flow meter to adjust or calibrate the respective feedwater flow nozzle-based signals. If the LEFM system or a portion of the system becomes inoperable, control room operators are promptly alerted by a Plant Monitoring System (PMS) alarm in the control room. Feedwater flow input to the core thermal power calculation would then be transferred to the feedwater flow nozzles in accordance with station procedures.

The electronics cabinet performs on-line, continuous monitoring of system parameters. Problems with the instrumentation or system malfunctions will result in PMS alarms and an indication of a change in the system status in the control room.

Each unit at PBAPS utilizes three LEFM flow meters, one in each of the three feedwater lines. The meters have three modes: NORMAL, MAINTENANCE, and FAIL. If an LEFM flow meter is in a status other than NORMAL, the uncertainty for that meter is increased. The total thermal power uncertainty increases with each LEFM flow meter that is in a status other than NORMAL.

The new TRM section 3.20 for the LEFM system, as provided in Attachment 3, is conservative and simple for operators to interpret and implement, and aligns with recent precedence for other industry MUR submittals. With any of the three LEFM flow meters in the MAINTENANCE mode but none in FAIL mode, TRM 3.20 will allow operation at the TPO power level of $\leq$ 4016 MWt for up to 72 hours. If all three LEFM flow meters are not in NORMAL mode at the end of 72 hours, power must be reduced to 4010 MWt. This power level is discussed in the Response to Criterion 6 below and is the power level calculated in Attachment 8 for the total thermal power uncertainty corresponding to all LEFM flow meters in the MAINTENANCE mode. If any of the LEFM flow meters are in the FAIL mode, the flow input of the failed LEFM flow meter(s) to the Core Thermal Power Calculation must be replaced with the associated feedwater flow nozzle input. The LEFM flow meter(s) in FAIL mode must then be restored to the NORMAL or MAINTENANCE mode within 72 hours or power must be reduced to pre-MUR power level of $\leq$ 3951 MWt.

The 72-hour allowed outage times (AOT) for the LEFM flow meter(s) prior to reducing to the intermediate power level of 4010 MWt or to 3951 MWt (CLTP) are acceptable because the existing feedwater flow nozzle-based signals will be calibrated to the last validated data from the LEFM system during this period. Any slight drift of the feedwater flow nozzle measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. This is conservative since the reactor will actually be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely and any significant sudden de-fouling would be detected by other plant parameters.

Regarding potential drift in the measurement of feedwater differential pressure across the flow nozzle, industry experience for similar Boiling Water Reactors (BWRs) shows that the instrument drift associated with feedwater flow measurements are insignificant over a 72-hour period. Table A-1 of Reference 8 indicates that the systematic error associated with feedwater flow nozzle differential pressure is approximately 1.0% over an operating cycle. Thus, over a 72-hour period, feedwater flow nozzle instrument drift would have an insignificant effect on the feedwater flow measurement.

<u>Criterion 2</u>

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response to Criterion 2

The PBAPS LEFM system installed instrumentation is representative of the LEFM system and is bounded by the analysis and assumptions set forth in Topical Report ER-80P (Reference 8).

The LEFM system has been highly reliable. The maintenance history of the LEFM system since January 2011 has been reviewed. On three occasions, repairs took more than 72 hours to return an LEFM flow meter from the MAINTENANCE mode to the NORMAL mode.

Recommended maintenance practices from the LEFM system vendor that have changed since original installation of the LEFM system have been appropriately incorporated for implementation at PBAPS. Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers are replaced as determined to be necessary by a review of the equipment's performance history by the LEFM system vendor.

The operational and maintenance history of these components shows that the system is reliable for feedwater flow measurement and thermal power calculations.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on the accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

This LEFM system uncertainty calculation methodology is based on EGC-accepted PBAPS plant setpoint methodology. The methodology used to calculate LEFM system uncertainty is described in the American Society of Mechanical Engineers PTC 19.1 methodology (Reference 12) and Caldon Engineering Reports ER-80P (Reference 8), as supplemented by ER-590 (Reference13) and by ER-160P (Reference 14).

Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (i.e., flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

As described in Section 3.3.2 above, the LEFM system was installed at PBAPS after the receipt of the initial MUR license amendment in 2002. Although it has not been credited in the safety analyses since implementation of the EPU license amendment, it remains the primary system to measure feedwater flow and provide input to the Core Thermal Power calculator. The calibration factors for the PBAPS LEFM flow meter spool pieces were established by tests of these spools at Alden Research Laboratory in May 2002. These included tests of a full-scale model of the PBAPS hydraulic geometry and tests in a straight pipe. The piping configuration at PBAPS remains bounded by the original LEFM flow meter installation and calibration assumptions as analyzed in the Cameron Topical Reports (References 8 and 9).

Criterion 5

Justification for continued operation at the pre-failure level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.

Response to Criterion 5

Justification for continued operation at the pre-failure power level for a pre-determined time and the actions taken in the event that time is exceeded (i.e., power reduction) is provided in the response to Criterion 1 above.

Criterion 6

A CheckPlus operating with a single failure is not the same as an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at a degraded uncertainty.

Response to Criterion 6

When the LEFM CheckPlus meter on any of the three PBAPS LEFM flow meters has only one of its two LEFM Check subsystems fully operational, resulting in that meter computing flow from just the remaining fully operational LEFM Check subsystem, that LEFM flow meter is considered in the MAINTENANCE mode. This status is indicated to operators on the PMS computer in the control room. The total thermal power uncertainties for the three LEFM flow meter conditions in the proposed TRM described in the Response to Criterion 1 above have been quantified on a plant-specific basis in Attachment 8 in accordance with the methodology of ER-157P (Reference 9). The table below provides the results of these calculations with the highest uncertainty for each condition shown:

LEFM FLOW METER CONDITION	TOTAL UNCERTAINTY	ASSOCIATED POWER LEVEL (MWt)
All in NORMAL	0.34%	4016
One or More in MAINTENANCE and None in FAIL	0.50%	4010

Criterion 7

An applicant with a comparable geometry can reference the above Section 3.2.1 finding [of the Final NRC Safety Evaluation for Caldon Topical Report ER-157P Rev 8 (Reference 11)] *to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.*

Response to Criterion 7

The NRC has determined in Reference 15 that for conditions in which the CheckPlus system is operating with one or more transducers out of service, the effect of downstream piping should be addressed if the separation distance from the meter transducers to the downstream piping change is less than five pipe diameters. At PBAPS, the LEFM flow meters are installed upstream of the feedwater flow nozzles, and the distance from meter transducers to downstream piping changes, i.e., venturi contraction, is greater than five pipe diameters in each feedwater line. Therefore, it is concluded that the downstream geometries for PBAPS do not have a significant influence on CheckPlus calibration.

Criterion 8

An applicant that requests an MUR with the upstream flow straightener configuration discussed in Section 3.2.2 [of the Final NRC Safety Evaluation for Caldon Topical Report ER-157P Rev 8 (Reference 11)] *should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17*. *Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener*.

Response to Criterion 8

The installed configuration of the PBAPS LEFM flow meters does not include an upstream flow straightener.

Criterion 9

An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of uncertainties or, alternatively, should ensure that *their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18*.

Response to Criterion 9

The PBAPS moisture separators and dryers deliver steam with low moisture content (i.e., ≤0.10 wt%). In the uncertainty calculation (Attachment 8), steam quality is not measured, but is taken as a constant moisture fraction of 0.0%. This conservatively increases the calculated core thermal power because the calculated value will not be reduced by the enthalpy lost to the moisture carryover. Thus, no additional uncertainty in steam enthalpy due to steam quality is calculated.

3.3.5 Deficiencies and Corrective Actions

Cameron, PBAPS and the other Licensees with LEFM systems actively participate in sharing operating experience. Cameron provides an analysis of LEFM system performance and reliability issues during their annual User's Group meeting. Prior to each meeting, Cameron collects from each user copies of condition reports related to the LEFM systems. Additionally, Cameron notifies all users of system related problems that could affect users using their Costumer Information Bulletin (CIB) process. The CIB topics are covered at the User Group meetings. PBAPS Site Engineering participates in the annual Cameron meetings and conducts periodic EGC fleet LEFM peer conference calls during which EGC fleet operating experience and issues are shared. Issues requiring action are tracked and addressed in the PBAPS corrective action program.

Problems with PBAPS plant instrumentation are documented in the PBAPS corrective action program and necessary corrective actions are identified and implemented. Deficiencies associated with the vendor's processes or equipment are reported to Cameron to support corrective actions.

3.3.6 Reactor Power Monitoring

PBAPS Unit 2 and Unit 3 have procedures that provide guidance for monitoring and controlling reactor power and ensuring that reactor power remains within the requirements of the operating license.

3.4 Evaluation of Changes to Licenses and Technical Specifications

The proposed changes to the TS previously described above in Section 2.0, "Description of Changes," are evaluated below.

Section 2.1, (change in RTP)

The proposed increase of 65 MWt (approximately 1.66%) in RTP in the operating license and TS definitions is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this license amendment request.

Section 2.2 Items 2 through 5, 8, 14, 16 and 19 (revised values for Thermal Power Monitoring Threshold)

PBAPS operates under the requirements of the stability Long-Term Solution (LTS) DSS-CD solution Option III. The OPRM system may only cause a scram when plant operation is in the Armed Region. Based on the approach described in Reference 1, Section 5.3.4, “Thermal-Hydraulic/Neutronic Stability,” for a power-uprated plant, the thermal limits monitoring threshold is scaled to a lower percent value to maintain the same absolute power/flow region boundaries.

Section 2.2 Items 6, 9, 13 and 15 (revised turbine scram bypass level)

Based on the guidelines in Reference 1, Section F.4.2.3, “Turbine First-Stage Pressure Signal Setpoint,” the value at which the turbine stop valve closure scram and turbine control valve fast closure scram are bypassed, in percent of RTP, is reduced by the ratio of the power increase. The value does not change with respect to absolute thermal power.

Section 2.2 Items 7 and 11 (OPRM Armed Region operability)

The OPRM is required to be operable above a power level set at 5% of rated power below the power boundary of the OPRM Armed Region defined by the thermal limits monitoring threshold (i.e., 22.6% - 5.0% = 17.6%).

Section 2.2 Items 10 and 12 (Allowable Value (AV) for Single and Two Loop APRM Flow Biased STP scram)

The proposed change to the nominal trip setpoints and AVs for the STP - Upscale function are based on the approach described in Reference 1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." The STP analytical limits (ALs) and AVs, for both TLO and SLO, are unchanged in units of absolute core thermal power versus recirculation drive flow. Because these values are expressed in percent of RTP, they decrease in proportion to the power uprate.

Section 2.2 Items 17 and 18 (change in reactor pressure limit for HPCI and RCIC testing)

The current PBAPS TS SR for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system pump flow surveillance tests (SR 3.5.1.8 and SR 3.5.3.3, respectively) require the test to be performed with a reactor pressure $\leq$ 1053 psig and $\geq$ 915 psig to confirm that the required pump flow rate can be achieved. As a result of the PBAPS Units 2 and 3 MUR LAR, the reactor pressure at rated MUR conditions will remain the same as CLTP conditions. To retain the reactor pressure the same at full power, the steam pressure at the main turbine stop valves must be lowered from 915 to approximately 910 psig to accommodate the increase in steam flow rate and the resulting higher steam pressure drop across the main steam lines and valves. The lower reactor pressure limit for performing the HPCI and RCIC system tests will also be changed from 915 to 910 psig to align with plant startup operations. This change will reduce challenges to the control of reactor pressure and reactivity when performing the surveillances during plant startup. A similar change was made following implementation of the EPU and was approved in Amendments 308 and 312, for Unit 2 and Unit 3, respectively (Reference 16).

An evaluation was performed at MUR conditions of the technical analysis of the similar change made for EPU conditions. The evaluation concluded the technical justification remains applicable to the change in the lower reactor pressure limit for performing the HPCI and RCIC system tests from 915 to 910 psig at MUR conditions. The current and proposed lower values of 915 psig and 910 psig, respectively, are consistent with the minimum EHC pressure setpoint at which reactor power can be increased without the need to adjust the EHC pressure setpoint during operation in Mode 1. Lowering the lower test pressure from 915 to 910 psig does not impact when the performance of the test is required. Neither the required HPCI and RCIC pump flow rates of 5000 gpm and 600 gpm, respectively, nor the pump discharge pressures are being changed. SRs 3.5.1.8 and 3.5.3.3 are intended to verify the operation of the HPCI and RCIC systems at the upper end of the reactor pressure operating range and such operation has been found to be safe, as authorized in PBAPS Units 2 and 3 Amendments 290/293, 293/296 and 308/312 (References 17, 5 and 16). Based on operating experience associated with the HPCI and RCIC system operations, there is margin in the turbine-pump systems that ensure that both HPCI and RCIC pumps will successfully pass SR 3.5.1.8 and SR 3.5.3.3 at the lower steam pressure of 910 psig. Lowering the pressure band at which the surveillance tests are performed provides a benefit by reducing the potential for a plant transient.

An evaluation of all of the % RTP values in the TS and TSB was performed and confirmed that all other % RTP values remain valid and are unchanged.

3.5 Additional Considerations

3.5.1 Summary of Analyses

The following is a summary of the analyses performed in support of these proposed changes, along with the results and a reference to the sections of Attachment 5 that provide further detail.

Topic	Conclusion	Attachment 5 Section
Normal plant operating conditions	MUR power uprate is accommodated by increasing core flow along previously established MELLLA Plus boundary lines.	Section 1
Reactor core and fuel performance	All fuel and core design limits are met.	Section 2
Reactor coolant and connected systems	Overpressure protection, fracture toughness, structural, and piping evaluations are acceptable. No increase in RPV overpressure because previous analyses accounted for ≥ 102% overpower, bounding TPO operation.	Section 3
Engineered safety features	Acceptable based on either plant-specific analyses or previous analyses at 102% of current licensed power	Section 4

Topic	Conclusion	Attachment 5 Section
Instrumentation and control	There are changes to some TS values as a result of TPO. No modifications to instruments are needed other than expected setpoint changes and replacement of some non-safety related instrumentation to provide adequate range for TPO.	Section 5
Electrical power and auxiliary systems	Electrical power and cooling water systems required for design basis events and Spent Fuel Pool Cooling system were previously analyzed at 102% CLTP thereby bounding TPO conditions Emergency operation at TPO is achieved by operating existing equipment at or below nameplate rating and within calculated BHP for ECCS pumps. The other auxiliary systems were evaluated at TPO conditions and are either not affected by TPO or remain adequate at TPO conditions.	Section 6
Power conversion systems	Power conversion systems are adequate without modification although there is the potential for a turbine-related hardware modification to take full advantage of the MUR uprate as discussed in section 3.5.3 below.	Section 7
Radwaste, radiation sources and radiation levels	The post-accident radiation levels were previously evaluated at 102% CLTP which bounds TPO and the TPO uprate was determined to have no significant effect on the plant or the habitability of the onsite emergency response facilities. Normal onsite and off-site radiation levels and the processing or liquid and solid radioactive waste are not significantly affected by TPO	Section 8
Reactor safety performance evaluations	Anticipated Operational Occurrences were previously evaluated at 102% CLTP at MELLLA+ conditions and bound the effect of the TPO uprate. Design basis accidents have either been evaluated at 102% CLTP which bounds the TPO power level or are not dependent on core thermal power. A plant-specific analysis of the limiting ATWS events was performed at 4018 MWt bounding the TPO. A plant-specific evaluation of Station Blackout was performed that confirmed continued compliance to 10 CFR 50.63 at TPO conditions.	Section 9
Other evaluations	Plant-specific evaluations of high and moderate energy line breaks, flow accelerated corrosion and conditions affecting environmental qualifications determined that the previous evaluations at CLTP conditions bound the consequences of any postulated transients at TPO conditions.	Section 10

3.5.2 Adverse Flow Effects

Industry experience has revealed that power uprate conditions can cause vibrations associated with acoustic resonance that can lead to steam dryer and main steam line (MSL) valve degradation. Monitoring for adverse flow affects was performed during EPU power ascension testing and during full EPU power operation and the vibration levels were found to be acceptable. Evaluations for flow-induced vibrations at MUR conditions were performed and concluded the vibration levels would remain acceptable as further discussed below.

The generic methods and assumptions of the Main Steam Isolation Valves (MSIVs) and Main Steam Line Flow Restrictors provided in the TLTR (Reference 1), Appendix J.2.3.7 have been evaluated and determined to be applicable to PBAPS. The requirements for the MSIVs remain unchanged for MUR uprate conditions. All safety and operational aspects of the MSIVs are within previous evaluations. See Attachment 5, Section 3.7, Main Steam Line Flow Restrictors, for details.

A plant specific evaluation to determine the effects of flow induced vibration on reactor internals at 110% rated core flow and at 102% of CLTP found that the vibrations of all safety-related reactor internal components are within acceptance criteria. See Attachment 5, Section 3.4, Flow Induced Vibration, for details.

Plant specific evaluations/stress reconciliations of reactor internals including core support structures and non-core support structures in support of TPO were conducted. The evaluations determined that all applicable loads remain unchanged or unaffected by TPO and that all RPV internals are within allowable limits. See Attachment 5, Section 3.3.2, Reactor Internals Structural Evaluation, for details.

The PBAPS Units 2 and 3 replacement steam dryers (RSDs) were analyzed using the same NRC approved methodologies used to structurally qualify the PBAPS steam dryers for EPU operating conditions. A high-cycle fatigue and ASME analysis of the RSDs has been completed utilizing main steam line data extrapolated to predicted MUR and MELLLA+ conditions. This analysis used the Westinghouse steam dryer main steam line acoustic methodology and the non-main steam line acoustic methodology. The results of the analysis verify the continued structural integrity of the PBAPS Units 2 and 3 steam dryers at predicted MUR and MELLLA+ conditions. See Attachment 10 for the detailed evaluation report.

A complete baseline visual inspection of the Unit 2 Replacement Steam Dryer was successfully performed during the 2016 Refueling Outage in accordance with RFOL Condition 2.C(15)(f). All observations were acceptable for the structural components and welds inspected.

3.5.3 Plant Modifications

No modifications are required other than some instrument setpoint changes, rescaling or replacements to accommodate TPO conditions. There is also the potential that turbine-related hardware may need to be modified to take full advantage of the MUR power uprate. Any such potential limitation would be an economic issue with no impact on the safe operation of the plant. See Attachment 5 Section 5.2.1, Pressure Control System, for details. These modifications will be made in accordance with 10 CFR 50.59, “Changes, tests, and experiments” and do not require NRC approval through this amendment request.

3.5.4 Instrument Setpoint Methodology

As described in Section 2.0, the only proposed changes to TS Allowable Values as a result of an MUR uprate are for the APRM Simulated Thermal Power – High for SLO and TLO operation. The determination of the AVs and Nominal Trip Setpoints (NTSPs) are based on the GEH setpoint methodology specified in NEDC-31336P-A, General Electric Instrument Setpoint Methodology, September 1996 (Proprietary). Each actual trip setting is established to preclude inadvertent initiation of the protective action, while assuring adequate allowances for instrument accuracy, calibration, drift and applicable normal and accident design basis events. See Attachment 5, Section 5.3 Technical Specification Instrument Setpoints, for details.

PBAPS previously adopted Technical Specification Task Force (TSTF) Traveler TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Revision 4, for the Average Power Range Monitor Simulated Thermal Power – High, Main Steam Line Flow – High, and Main Steam Line pressure – Low as part of the EPU License Amendments 293 and 296 for Units 2 and 3 (Reference 5). These amendments also added Notes (e) and (f) to TS Table 3.3.1.1-1 in accordance with TSTF-493.

3.5.5 Grid Studies

Grid studies by PJM Interconnection, the regional transmission organization, and PECO, the local electrical power distribution utility, that support the proposed uprate are included in Attachment 13.

PJM completed a study to assess the impact of the uprate on system stability. The analysis assumed a 1392 MWe output for the PBAPS Units 2 and 3 main generators, which bounds the highest expected electrical output under uprate conditions, and a light load base case based on 2018 projections, modified to include applicable queue projects. A range of contingencies necessary to assess compliance with North American Electric Reliability Corporation (NERC), PJM and other applicable criteria were evaluated. For all cases studied, transient stability is maintained with all oscillations stabilized within 20 seconds and voltage levels returned to acceptable levels following the fault clearance. Hence, no transient stability issues were identified.

PECO Transmission Planning completed a study to determine if the capacity and capability of the preferred power supply ensures the design and licensing basis for PBAPS Units 2 and 3 under MUR conditions. Four load flow cases representative of 90/10 and 50/50[1] summer peak loads, 75% of the 50/50 peak, and a spring light load case were used for the simulations. The load profiles for these cases are based on actual load readings for summer peak, winter peak, and spring light load conditions. In order to further stress the transmission system, power flow simulations were also performed with an additional 1000 MW flowing across the PJM eastern interface in the 50/50 peak load and 75% of 50/50 peak load cases. The uprate study verified the

[1] A 90/10 forecast provides a peak load projection with only a 10% probability that the actual peak will be higher than the level forecasted in that year. A 50/50 load forecast provides a peak load projection that has an equal probability of being higher or lower than the peak load that actually occurs in that year.

transmission system's capability to maintain the post-trip voltage drops at values less than those required for operability of the safety busses.

On the basis of the PJM and PECO studies, EGC has determined that the MUR power uprate will have no significant effect on grid stability or reliability and no modifications to the transmission system are required.

3.5.6 Operator Training, Human Factors, and Procedures

Because PBAPS had implemented and installed the LEFM system in both units by 2003, MUR implementation at PBAPS will only have a minor impact on human factors in the areas of operating procedure changes, operator training and operator human performance. The PBAPS simulator will be modified to reflect any changes made to the main control room as part of the design change process with established PBAPS certification procedures and in accordance with ANSI/ANS-3.5 1998, Nuclear Power Plant Simulators for Use in Operator Training and Evaluation. Operator training for changes to control room interfaces, alarms, and indications will be accomplished prior to implementation of the proposed changes in accordance with the plant training and simulator program.

3.5.7 Testing

Plant testing for the proposed changes will be completed as described in Attachment 5, Section 10.4, "Testing."

4.0 REGULATORY EVALUATION

4.1. Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix K, "ECCS Evaluation Models," requires that emergency core cooling system evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 1, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

10 CFR 50, Appendix K does not permit licensees to utilize a lower uncertainty and increase thermal power without NRC approval. 10 CFR 50.90 requires that licensees desiring to amend an operating license file an amendment with the NRC.

RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," provides criteria for the content of license amendment requests involving power uprates based on measurement uncertainty recapture.

This application is consistent with the requirements and criteria described in 10 CFR 50, Appendix K, 10 CFR 50.90, and the guidelines of RIS 2002-03.

4.2. Precedent

The following facilities have recently received NRC approval for power uprates based on use of the LEFM CheckPlus system.

Facility	Amendment Nos.	Approval Date	Applicability to PBAPS MUR
Catawba	281	April 29, 2016	Smaller uncertainty (0.29%)
Fermi 2	196	February 10, 2014	Similar BWR with comparable uncertainty (0.361%)
Byron 1 and 2	181	February 7, 2014	Comparable uncertainty (0.345%)
Braidwood 1 and 2	174	February 7, 2014	Comparable uncertainty (0.345%)
McGuire 1 and 2	269/249	May 6, 2013	Smaller uncertainty (0.29%)
Shearon Harris	139	May 30, 2012	Comparable uncertainty (0.34%) and similar LEFM inoperability strategy
Limerick 1 and 2	201/163	April 8, 2011	Similar BWR with comparable uncertainty (0.347%)
River Bend	129	January 31, 2003	Similar BWR with smaller uncertainty (0.30%)

4.3. No Significant Hazards Consideration

In accordance with 10 CFR 50.90, “Application for Amendment of License, Construction Permit, or Early Site Permit,” and 10 CFR 50, Appendix K, “ECCS Evaluation Model,” Exelon Generation Company, LLC (EGC) is proposing that Renewed Facility Operating Licenses Nos. DPR 44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3, respectively, be amended to reflect an increase of 65 MWt (approximately 1.66%) in rated thermal power (RTP) from 3951 megawatts thermal (MWt) to 4016 MWt for each unit. The increase in RTP is achieved by use of the Cameron International (formerly Caldon, Inc.) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation that have been installed in both PBAPS Units 2 and 3 since 2003. The CheckPlus™ LEFM provides improved feedwater flow measurement accuracy and thus improved operational power level certainty. EGC has evaluated whether or not a significant hazards consideration is involved with the proposed changes in accordance with the three standards set forth in 10 CFR 50.92, “Issuance of Amendment,” as discussed below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No, the proposed increase in power level does not significantly increase the probability or consequences of an accident previously evaluated.

   The proposed changes do not affect system design or operation and thus do not create any new accident initiators or increase the probability of an accident previously evaluated. All accident mitigation systems will function as designed, and all performance requirements for these systems have been evaluated and were found acceptable.

   The Nuclear Steam Supply System (NSSS) components (e.g., reactor vessel, reactor internals, control rod drive housings, piping and supports, and recirculation pumps) remain within their applicable structural limits and will continue to perform their intended design functions during normal and accident conditions. Thus, there is no increase in the probability of a structural failure of these components.

   The balance of plant systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a failure of these components. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level. Because the integrity of the plant will not be affected by operation at the uprated condition, EGC has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions.

   All safety analyses have either been performed at 102% of Current Licensed Thermal Power (CLTP) and therefore bound the proposed uprate or have been subject to plant-specific analyses at a power level equal to or greater than the proposed uprate. The results demonstrate that acceptance criteria of the applicable analyses continue to be met at the uprated conditions. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No, the proposed increase in power level does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety-related system. No new equipment or procedure changes are involved that could add new accident initiators.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No, the proposed increase in power level does not involve a significant reduction in a margin of safety.

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier, and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed or approved by the Nuclear Regulatory Commission, or that are in compliance with regulatory review guidance and standards.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.4. Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusions or otherwise not requiring environmental review," addresses requirements for submitting environmental assessments as part of licensing actions. 10 CFR 51.22, paragraph (c)(9) states that a categorical exclusion applies for Part 50 license amendments that meet the following criteria:

i. No significant hazards consideration (as defined in 10 CFR 50.92(c));

ii. No significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and

iii. No significant increase in individual or cumulative occupational radiation exposure.

The proposed changes do not involve a significant hazards consideration. The reviews and evaluations performed to support the proposed uprate conditions concluded that all systems will function as designed, and all performance requirements for these systems have been evaluated and found acceptable. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Operation at the uprated power condition does not involve a significant reduction in a margin of safety.

No significant changes in types or amounts of effluents released into the environment will occur as a result of the power uprate. The Pennsylvania Department of Environmental Protection (PDEP) National Pollutant Discharge Elimination System (NPDES) permit provides the effluent limitations and monitoring requirements for wastewater at the site.

There is no significant increase in individual or cumulative occupational radiation exposure. Evaluations of projected radiation exposure concluded that normal operation radiation levels increase slightly for the proposed uprate, but that occupational exposure is controlled by the plant radiation protection program and is maintained well within values required by regulations.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NEDC 32938P-A, “Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” dated May 2003

2. Letter from NRC to George A. Hunger, “Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 2 (TAC No. M86826), dated October 18, 1994

3. Letter from NRC to George A. Hunger, “Revised Maximum Authorized Thermal Power Limit, Peach Bottom Atomic Power Station, Unit No. 3 (TAC No. M86827), dated July 18, 1995

4. Letter from NRC to John L. Skolds, “Peach Bottom Atomic Power Station, Units 2 And 3 – Issuance of Amendment Re: 1.62% Increase In Licensed Power Level (TAC Nos. MB5192 and MB5193),” dated November 22, 2002

5. Letter from NRC to Michael J. Pacillo, “Peach Bottom Atomic Power Station, Units 2 and 3- Issuance of Amendments Re: Extended Power Uprate (TAC NOS. ME9631 and ME9632),” dated August 25, 2014.

6. Letter from NRC to Bryan C. Hanson, “Peach Bottom Atomic Power Station, Units 2 and 3- Issuance of Amendments Re: Maximum Extended Load Line Limit Analysis Plus (CAC NOS. MF4760 and MF4761),” dated March 21, 2016.

7. NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002

8. Caldon Topical Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System,” Rev. 0, dated March 1997

9. Caldon Topical Report ER-157P, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM √ ™ or an LEFM CheckPlus ™ System,” Rev. 8, dated October 2001

10. Letter from NRC to C. Lance Terry, “Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System,' ” dated March 8, 1999

11. Letter from Thomas B. Blount (USNRC) to Ernest Hauser (Cameron), “Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, ‘Caldon Ultrasonics Engineering Report ER-157P, Supplement to Topical Report ER-80P: Basis for Power Uprate with the LEFM Check or CheckPlus System,’ (TAC No. ME1321),” dated August 16, 2010 (ML102160663)

12. ASME PTC 19.1-1998, “Test Uncertainty, Instruments and Apparatus,” American Society of Mechanical Engineers, 1998

13. Cameron Engineering Report ER-590, “Effect of Random and Coherent Noise on LEFM CheckPlus Systems, Revision 2,” June 2007

14. Caldon Engineering Report ER-160P, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM CheckPlus System” Revision 0, dated May 2000

15. Letter from John P. Boska, NRC to Steven D, Capps, “McGuire Nuclear Stations Units 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Power Uprate (TAC NOS. ME8213 AND ME8214), dated May 16, 2013 (ADAMS Accession No. ML13073A041)

16. NRC letter to Exelon, “Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments RE: Surveillance Requirements for High Pressure Coolant Injection System and Reactor Core Isolation Cooling System”, dated July 5, 2016 (ADAMS Accession No. ML16159A148)

17. NRC letter to Exelon, “Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendments Safety Relief Valve and Safety Valve Lift Setpoint Tolerance (TAC Nos. MF1970 and MF1971)”, dated May 5, 2014 (ADAMS Accession No. ML14079A102)

18. NRC Regulatory Guide 1.49, “Power Levels of Nuclear Power Plants,” Revision 1, December 1973

ATTACHMENT 2

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

Markup of Proposed Operating License and Technical Specifications Pages for Units 2 and 3

Renewed Facility Operating License Pages

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Technical Specification Pages

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3.3-31a	3.3-31a
3.3-31b	3.3-31b
3.3-31c	3.3-31c
3.4-7	3.4-7
3.5-6	3.5-6
3.5-13	3.5-13
3.7-12	3.7-12

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit 2, at steady state reactor core power levels not in excess of ~~3951~~ megawatts thermal. 4016

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 310, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 281 and modified by Amendment No. 301.

(4) Fire Protection

The Exelon Generation Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report (SER) dated May 23, 1979, and Supplements dated August 14, September 15, October 10 and November 24, 1980, and in the NRC SERs dated September 16, 1993, and August 24, 1994, subject to the following provision:

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

1.1 Definitions

PHYSICS TESTS (continued)	b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3951~~ 4016 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.
RECENTLY IRRADIATED FUEL	RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ ~~23%~~ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ 22.6% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ 22.6% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

22.6%

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more automatic Functions with RPS trip capability not maintained. <u>OR</u> Two or more manual Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < ~~26.7%~~ 26.3% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR.	Immediately
	<u>AND</u>		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	<u>AND</u>		
	I.3	Initiate action to submit an OPRM report in accordance with Specification 5.6.8.	Immediately
J. Required Action and associated Completion Time of Condition I not met.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	<u>AND</u>		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>		
	J.3	-------- NOTE -------- LCO 3.0.4 is not applicable. --------------------- Restore required channel to OPERABLE.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to < ~~18%~~ RTP. 17.6%	4 hours

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTES-----------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ ~~23%~~ 22.6% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~23%~~ 22.6% RTP.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.12	-------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. -- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.13	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.14	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.15	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Wide Range Neutron Monitors					
a. Period-Short	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
b. Inop	2	3	G	SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5[(a)]	3	H	SR 3.3.1.1.6 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux-High (Setdown)	2	3[(c)]	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 15.0% RTP
b. Simulated Thermal Power-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12[(e),(f)]	~~≤ 0.61 W + 67.1% RTP~~ ≤ 0.60W + 65.9% RTP [(b)(g)] and ≤ 118.0% RTP
c. Neutron Flux-High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 119.7% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	NA
f. OPRM Upscale	≥ ~~18%~~ 17.6%[(h)] RTP	3[(c)]	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.55 (W − ΔW) + 61.5% RTP~~ 0.54 (W - ΔW) + 60.3% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) Deleted

(e) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(f) The instrument channel set point shall be reset to a value that is within the Leave Alone Zone (LAZ) around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found tolerance and LAZ apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP methodologies used to determine the as-found tolerance and the LAZ are specified in the Bases associated with the specified function.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.	Reactor Pressure —High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 1085.0 psig
4.	Reactor Vessel Water Level—Low (Level 3)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 1.0 inches
5.	Main Steam Isolation Valve —Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
6.	Drywell Pressure — High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 2.0 psig
7.	Scram Discharge Volume Water Level —High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 50.0 gallons
		5[(a)]	2	H	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 50.0 gallons
8.	Turbine Stop Valve —Closure	≥ ~~26.7%~~ RTP 26.3%	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure —Low	≥ ~~26.7%~~ RTP 26.3%	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 500.0 psig
10.	Turbine Condenser —Low Vacuum	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 23.0 inches Hg vacuum
11.	Deleted					
12.	Reactor Mode Switch — Shutdown Position	1,2	1	G	SR 3.3.1.1.14 SR 3.3.1.1.17	NA
		5[(a)]	1	H	SR 3.3.1.1.14 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels per trip system of the Digital Feedwater Control System (DFCS) high water level trip instrumentation Function shall be OPERABLE.

22.6%

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ RTP.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DFCS high water level trip channels inoperable.	A.1 Place channel in trip.	72 hours
B. DFCS high water level trip capability not maintained.	B.1 Restore DFCS high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 --------NOTE-------- Only applicable if inoperable channel is the result of inoperable feedwater pump turbine or main turbine stop valve. -------------------- Remove affected feedwater pump(s) and main turbine valve(s) from service.	4 hours
	<u>OR</u>	
	C.2 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

3.3 INSTRUMENTATION

3.3.4.2 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.2

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)—Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure—Low.

OR

b. The following limits are made applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for inoperable EOC-RPT as specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR; and
3. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~26.7%~~ RTP.

26.3%

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	OR	
	A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained.	B.1	Restore EOC-RPT trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	---------NOTE-------- Only applicable if inoperable channel is the result of an inoperable RPT breaker. ---------------------	
		Remove the affected recirculation pump from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < ~~26.7%~~ 26.3% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV—Closure: ≤ 10% closed; and TCV Fast Closure, Trip Oil Pressure—Low: ≥ 500 psig.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.3	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.4	Verify TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.5	-------------------NOTE------------------- Breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6. -- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.6	Determine RPT breaker interruption time.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > ~~23%~~ 22.6% RTP. --- Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop: a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns. b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. c. Each jet pump flow differs by ≤ 10% from established patterns.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.8	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1053 and ≥ ~~915~~ 910 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.9	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.10	-------------------NOTE------------------- Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.3	-------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- 910 Verify, with reactor pressure ≤ 1053 psig and ≥ ~~915~~ psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.4	-------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.

(continued)

3.7 PLANT SYSTEMS

3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR),” limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ 22.6% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

(5) Exelon Generation Company, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Limerick Generating Station, Units 1 and 2.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the Peach Bottom Atomic Power Station, Unit No. 3, at steady state reactor core power levels not in excess of ~~3951~~ 4016 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 314, are hereby incorporated in the license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Peach Bottom Atomic Power Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 283 and modified by Amendment No. 304.

[1] The Training and Qualification Plan and Safeguards Contingency Plan and Appendices to the Security Plan.

1.1 Definitions

PHYSICS TESTS (continued)

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3951~~ 4016 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from the opening of the sensor contact up to and including the opening of the trip actuator contacts.

RECENTLY IRRADIATED FUEL

RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 312 hours. This 312-hour time period may be reduced to 24 hours if secondary containment hatches H2, H21, H22 and H34 are closed.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

a. The reactor is xenon free;

b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

22.6%

THERMAL POWER shall be ≤ ~~23%~~ RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

22.6%

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP AND In accordance with the Surveillance Frequency Control Program.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

22.6%

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP AND In accordance with the Surveillance Frequency Control Program.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ 22.6% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~23%~~ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more automatic Functions with RPS trip capability not maintained. <u>OR</u> Two or more manual Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~26.7%~~ RTP. 26.3%	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR.	Immediately
	<u>AND</u>		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	<u>AND</u>		
	I.3	Initiate action to submit an OPRM report in accordance with Specification 5.6.8.	Immediately
J. Required Action and associated Completion Time of Condition I not met.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	<u>AND</u>		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>		
	J.3	------ NOTE ---------- LCO 3.0.4 is not applicable. --------------------- Restore required channel to OPERABLE.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to < ~~18%~~ RTP. 17.6%	4 hours

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTES-------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ ~~23%~~ 22.6% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~23%~~ 22.6% RTP.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.12	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. --- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.13	Verify Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ ~~26.7%~~ RTP. 26.3%	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.14	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.1.1.15	Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program.

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Wide Range Neutron Monitors					
a. Period-Short	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 13 seconds
b. Inop	2	3	G	SR 3.3.1.1.5 SR 3.3.1.1.17	NA
	5(a)	3	H	SR 3.3.1.1.6 SR 3.3.1.1.17	NA
2. Average Power Range Monitors					
a. Neutron Flux-High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 15.0% RTP
b. Simulated Thermal Power-High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12(e),(f)	≤0.60W + 65.9% RTP ≤ ~~0.61 W + 67.1% RTP~~(b)(g) and ≤ 118.0% RTP
c. Neutron Flux-High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	≤ 119.7% RTP
d. Inop	1,2	3(c)	G	SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.11 SR 3.3.1.1.17 SR 3.3.1.1.18	NA
f. OPRM Upscale	17.6% ≥~~18%~~(h) RTP	3(c)	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12	NA

(continued)

0.54 (W - ΔW) + 60.3% RTP

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.55 (W - ΔW) + 61.5% RTP~~ when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) Deleted

(e) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(f) The instrument channel set point shall be reset to a value that is within the Leave Alone Zone (LAZ) around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided the as-found tolerance and LAZ apply to the actual setpoint implemented in the Surveillance procedures to confirm channel performance. The NTSP methodologies used to determine the as-found tolerance and the LAZ are specified in the Bases associated with the specified function.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION		APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3.	Reactor Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 1085.0 psig
4.	Reactor Vessel Water Level – Low (Level 3)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 1.0 inches
5.	Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
6.	Drywell Pressure – High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 2.0 psig
7.	Scram Discharge Volume Water Level – High	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 50.0 gallons
		5(a)	2	H	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 50.0 gallons
8.	Turbine Stop Valve – Closure	≥ ~~26.7%~~ RTP 26.3%	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≤ 10% closed
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ ~~26.7%~~ RTP 26.3%	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 500.0 psig
10.	Turbine Condenser – Low Vacuum	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	≥ 23.0 inches Hg vacuum
11.	Deleted					
12.	Reactor Mode Switch – Shutdown Position	1,2	1	G	SR 3.3.1.1.14 SR 3.3.1.1.17	NA
		5(a)	1	H	SR 3.3.1.1.14 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Two channels per trip system of the Digital Feedwater Control System (DFCS) high water level trip instrumentation Function shall be OPERABLE.

22.6%

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ RTP.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DFCS high water level trip channels inoperable.	A.1 Place channel in trip.	72 hours
B. DFCS high water level trip capability not maintained.	B.1 Restore DFCS high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 --------NOTE--------- Only applicable if inoperable channel is the result of inoperable feedwater pump turbine or main turbine stop valve. -------------------- Remove affected feedwater pump(s) and main turbine valve(s) from service.	 4 hours
	<u>OR</u> 22.6% C.2 Reduce THERMAL POWER to < ~~23%~~ RTP.	 4 hours

3.3 INSTRUMENTATION

3.3.4.2 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.2

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)—Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure—Low.

<u>OR</u>

b. The following limits are made applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for inoperable EOC-RPT as specified in the COLR;

2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR; and

3. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for inoperable EOC-RPT as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	<u>OR</u>	
	A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained.	B.1	Restore EOC-RPT trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	--------NOTE--------- Only applicable if inoperable channel is the result of an inoperable RPT breaker. -------------------- Remove the affected recirculation pump from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < ~~26.7%~~ 26.3% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

-------------------------------------NOTE-------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV—Closure: ≤ 10% closed; and TCV Fast Closure, Trip Oil Pressure—Low: ≥ 500 psig.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.3	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.4	Verify TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~26.7%~~ RTP. 26.3%	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.5	------------------NOTE------------------ Breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6. -- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.3.4.2.6	Determine RPT breaker interruption time.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > ~~23%~~ 22.6% RTP. --- Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop: a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns. b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. c. Each jet pump flow differs by ≤ 10% from established patterns.	In accordance with the Surveillance Frequency Control Program.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1053 and ≥ 910 ~~915~~ psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.9	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 175 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.1.10	-------------------NOTE------------------- Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.3	-------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 1053 psig and ≥ ~~915~~ 910 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	 In accordance with the Surveillance Frequency Control Program.
SR 3.5.3.4	-------------------NOTE-------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with reactor pressure ≤ 175 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	 In accordance with the Surveillance Frequency Control Program.

(continued)

3.7 PLANT SYSTEMS

3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~23%~~ 22.6% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < ~~23%~~ 22.6% RTP.	4 hours

ATTACHMENT 3

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

<u>Markup of Proposed Technical Specifications Bases and Technical Requirements Manual Pages</u>

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B 3.2-4	B 3.3-60	B 3.2-4	B 3.3-61
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B 3.3-12b	B 3.5-14	B 3.3-12b	B 3.5-14
B 3.3-18	B 3.5-28	B 3.3-18	B 3.5-28
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3.20-1	3.20-1
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B 3.20-1	B 3.20-1
B 3.20-2	B 3.20-2
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BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of ~~23%~~ RTP for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 4.

22.6%

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

LCO (continued)

With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative factor.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are assumed to occur at high power levels. Design calculations (Ref. 6) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the wide range neutron monitor period-short scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < ~~23%~~ 22.6% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. The

(continued)

BASES

ACTIONS

B.1 (continued)

allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ 22.6% RTP and then periodically thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ 22.6% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single Loop Operation," May 1980.

6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

7. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

8. Deleted

9. NEDO-30130-A, "Steady State Nuclear Methods," April 1985.

(continued)

BASES

REFERENCES (continued)

10. Deleted

11. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

12. Peach Bottom Unit 2 Core Operating Limits Report (COLR).

13. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 10) to analyze slow flow runout transients. The flow dependent operating limit, $MCPR_f$, is evaluated based on a single recirculation pump flow runout event (Ref. 9).

Power dependent MCPR limits ($MCPR_p$) are determined by the codes used to evaluate transients as described in Reference 2. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between ~~23%~~ 22.6% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below ~~23%~~ 22.6% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below ~~23%~~ 22.6% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at ~~23%~~ 22.6% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and

(continued)

BASES

APPLICABILITY (continued)

flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to ~~23%~~ 22.6% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the wide range neutron monitor period-short function provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < ~~23%~~ 22.6% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ 22.6% RTP and periodically thereafter. It is compared to the specified limits

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

in the COLR (Ref. 12) to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ 22.6% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

3. UFSAR, Chapter 3.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 14.

6. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single Loop Operation," May 1980.

(continued)

BASES

REFERENCES (continued)

7. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

8. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

9. NEDC-32428P, "Peach Bottom Atomic Power Station Unit 2 Cycle 11 ARTS Thermal Limits Analyses," December 1994.

10. NEDO-30130-A, "Steady State Nuclear Methods," April 1985.

11. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

12. Peach Bottom Unit 2 Core Operating Limits Report (COLR).

13. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES (continued)

includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking.

Power-dependent and flow-dependent LHGR adjustment factors may also be provided per Reference 1 to ensure that fuel design limits are not exceeded due to the occurrence of a postulated transient event during operation at off-rated (less than 100%) reactor power or core flow conditions. These adjustment factors are applied, if required, per the COLR and decrease the allowable LHGR value.

Additionally, for single recirculation loop operation, an LHGR multiplier may be provided per Reference 1. This multiplier is applied per the COLR and decreases the allowable LHGR value. This additional margin may be necessary during SLO to account for the conservative analysis assumption of an earlier departure from nucleate boiling with only one recirculation loop available.

The LHGR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

22.6%

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < ~~23%~~ RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ ~~23%~~ RTP.

22.6%

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The

(continued)

BASES

ACTIONS

A.1 (continued)

2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < ~~23%~~ 22.6% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < ~~23%~~ 22.6% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

22.6%

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ RTP and periodically thereafter. It is compared to the specified limits in the COLR (Ref. 10) to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

22.6%

REFERENCES

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single-Loop Operation," May 1980.

6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvements Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

7. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

8. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

9. NUREG-0800, Section 4.2, Subsection II.A.2(g), Revision 2, July 1981.

10. Peach Bottom Unit 2 Core Operating Limits Report (COLR).

11. G-080-VC-400, “Peach Bottom Atomic Power Station Units 2 & 3 GNF2 ECCS-LOCA Evaluation,” GE Hitachi Nuclear Energy, 0000-0100-8531-R1, March 2011.

12. G-080-VC-272, “Peach Bottom Atomic Power Station ECCS-LOCA Evaluation for GE14,” General Electric Company, GENE-J11-03716-09-02P, July 2000.

13. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-High (Setdown) (continued)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux-High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux-High (Setdown) Function will provide a secondary scram to the Wide Range Neutron Monitor Period-Short Function because of the relative setpoints. At higher power levels, it is possible that the Average Power Range Monitor Neutron Flux-High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux-High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed ~~23%~~ 22.6% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < ~~23%~~ 22.6% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < ~~23%~~ 22.6% RTP.

The Average Power Range Monitor Neutron Flux-High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux-High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power-High

The Average Power Range Monitor Simulated Thermal Power-High Function monitors average neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux-High Function Allowable Value. A note is included, applicable when the plant is in single recirculation loop operation per LCO 3.4.1, which requires the flow value, used in the Allowable Value equation, be reduced by ΔW. The value of ΔW

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.b. Average Power Range Monitor Simulated Thermal Power-High</u> (continued)

is established to conservatively bound the inaccuracy created in the core flow/drive flow correlation due to back flow in the jet pumps associated with the inactive recirculation loop. The Allowable Value thus maintains thermal margins essentially unchanged from those for two loop operation. The value of ΔW is plant specific and is defined in plant procedures. The Allowable Value equation for single loop operation is only valid for flows down to W = ΔW; the Allowable Value does not go below ~~61.5%~~ 60.3% RTP. This is acceptable because back flow in the inactive recirculation loop is only evident with drive flows of approximately 35% or greater (Reference 19). The Nominal Trip Setpoint (NTSP) and the as-found and as-left tolerances (Leave Alone Zone) were determined in accordance with Reference 10.

The Average Power Range Monitor Simulated Thermal Power-High Function is not specifically credited in the safety analysis but is intended to provide an additional margin of protection from transient induced fuel damage during operation where recirculation flow is reduced to below the minimum required for rated power operation. The Average Power Range Monitor Simulated Thermal Power-High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Neutron Flux-High Function will provide a scram signal before the Average Power Range Monitor Simulated Thermal Power-High Function setpoint is exceeded.

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function. The APRM flow processing logic is considered inoperable whenever it cannot deliver a flow signal less than or equal to actual Recirculation flow conditions for all steady state and transient reactor conditions while in Mode 1. Reduced or Downscale flow conditions due to planned maintenance or testing activities during derated plant conditions (i.e. end of cycle coast down) will result in conservative setpoints for the APRM Simulated Thermal Power-High function, thus maintaining that function operable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

Reference 22 describes the Detect and Suppress-Confirmation Density (DSS-CD) long-term stability solution and the licensing basis Confirmation Density Algorithm (CDA). Reference 22 also describes the DSS-CD Armed Region and the three additional algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (PBDA), the amplitude based algorithm (ABA), and the growth rate algorithm (GRA). All four algorithms are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the CDA. The remaining three algorithms provide defense-in-depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY is based only on the CDA.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into cells for evaluation by the OPRM algorithms.

DSS-CD operability requires at least 8 responsive OPRM cells per channel. The DSS-CD software includes a self-check for the responsive OPRM cells; therefore, no SR is necessary.

The OPRM Upscale Function is required to be OPERABLE when the plant is ≥ ~~18%~~ 17.6% RTP, which is established as a power level that is greater than or equal to 5% below the lower boundary of the Armed Region. This requirement is designed to encompass the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The OPRM Upscale Function is automatically trip-enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ ~~23%~~ 22.6% RTP corresponding to the MCPR monitoring threshold and reactor recirculation drive flow, is less than 75% of rated flow. This region is the OPRM Armed Region. Note (h) allows for entry into the DSS-CD Armed Region without automatic arming of DSS-CD prior to completely passing through the DSS-CD Armed Region during the first startup and the first shutdown following DSS-CD implementation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.f. Oscillation Power Range Monitor (OPRM) Upscale</u> (continued)

As described in Reference 22 and 24, the RTP values for the OPRM Upscale Function to be OPERABLE (≥ ~~18%~~ 17.6% RTP) and for the OPRM Upscale Function to be auto-enabled (≥ ~~23%~~ 22.6% RTP) are sufficiently conservative for protection of the plant against thermal-hydraulic instabilities. The basis for the 5% RTP difference between the OPRM Upscale OPERABLE (~~18%~~ 17.6% RTP) and OPRM Upscale auto-enable value (~~23%~~ 22.6% RTP) is to ensure that no credible event, e.g., loss of feed water heating, could result in a plant power excursion where an inoperable OPRM channel entered into the OPRM Armed Region. Peach Bottom plant specific analyses performed at these low power levels (Reference 24) have demonstrated that any power excursion resulting from credible events is bounded by 5% RTP. In addition, both the core-wide and channel decay ratios at the OPRM Upscale auto-enabled values are extremely low as documented in Reference 22, which demonstrates the low possibility of thermal-hydraulic instabilities at low power and confirms the conservatisms in the OPRM Upscale Function auto-enable RTP value. The conservatisms in the determination of the values for OPRM Upscale Function OPERABLE and the OPRM Upscale Function auto enabled sufficiently compensate for possible inaccuracy of the APRM simulated thermal power signal versus actual core thermal power at power levels < ~~23%~~ 22.6% RTP. Therefore, there is no need to perform any calibration of the APRM simulated thermal power signal to calculated power with RTP < ~~23%~~ 22.6% in order to determine the OPRM Upscale Function OPERABLE.

If any OPRM auto-enable setpoint is in a non-conservative condition, i.e., the OPRM Upscale is not auto-enabled with RTP ≥ ~~23%~~ 22.6% and reactor recirculation drive flow ≤ 75% of rated, the associated channel is considered inoperable for the OPRM Upscale function. Alternatively, the auto-enable setpoint may be adjusted to place the channel in a conservative condition (armed). If placed in the armed condition, the channel is considered OPERABLE.

Note (h) reflects the need for plant data collection in order to test the DSS-CD equipment. Testing the DSS-CD equipment ensures its proper operation and prevents spurious reactor trips. Entry into the DSS-CD Armed Region without automatic arming of DSS-CD during this initial testing phase also allows for changes in plant operations to address maintenance or other operational needs. However, during this initial testing period, the OPRM Upscale Function is OPERABLE and DSS-CD operability and capability to automatically arm shall be maintained at recirculation drive flow rates above the DSS-CD Armed Region flow boundary.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL POWER ≥ ~~27.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This Function is not required when THERMAL POWER is < ~~26.7%~~ 26.3% RTP since the Reactor Pressure—High and the Average Power Range Monitor Scram Clamp Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7 and the generator load rejection with bypass failure event. For these events, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the relayed emergency trip supply oil pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This Function is not required when THERMAL POWER is < ~~26.7%~~ 26.3% RTP, since the Reactor Pressure—High and the Average Power Range Monitor Scram Clamp Functions are adequate to maintain the necessary safety margins.

10. Turbine Condenser—Low Vacuum

The Turbine Condenser—Low Vacuum Function protects the integrity of the main condenser by scramming the reactor and thereby decreasing the severity of the low condenser vacuum transient on the condenser. This function also ensures integrity of the reactor due to loss of its normal heat sink. The reactor scram on a Turbine Condenser—Low Vacuum signal will occur prior to a reactor scram from a Turbine Stop Valve—Closure signal. This function is not specifically credited in any accident analysis but is being retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

(continued)

BASES

ACTIONS (continued)

J.3

BSP is a temporary means for protection against thermal-hydraulic instability events. An extended period of inoperability without automatic trip capability is not justified. Consequently, the required channels are required to be restored to OPERABLE status within 120 days.

Based on engineering judgment, the likelihood of an instability event that could not be adequately handled by the use of the BSP Regions (See Action J.1) and the BSP Boundary (See Action J.2) during a 120-day period is negligibly small. The 120-day period is intended to allow for resolution of a variety of equipment problems (e.g., design changes, extensive analysis, or other unforeseen circumstances). This action is not intended to be used for operational convenience. Correction of most equipment failures or inoperabilities is expected to normally be accomplished within the completion times allowed for Actions for Conditions A and I.

A Note is provided to indicate that LCO 3.0.4 is not applicable. The intent of the note is to allow plant startup while operating within the 120-day Completion Time for Required Action J.3. The primary purpose of this exclusion is to allow an orderly completion of design and verification activities, in the event of a required design change, without undue impact on plant operation.

K.1

17.6%

If the required channels are not restored to OPERABLE status and the Required Actions of J are not met within the associated Completion Times, then the plant must be placed in an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to less than ~~18%~~ RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the specified operating power level from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

A restriction to satisfying this SR when < ~~23%~~ 22.6% RTP is provided that requires the SR to be met only at ≥ ~~23%~~ 22.6% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < ~~23%~~ 22.6% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ ~~23%~~ 22.6% RTP, the Surveillance is required to have been satisfactorily performed in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above ~~23%~~ 22.6% if the Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding ~~23%~~ 22.6% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

(Not Used.)

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.1.5 and SR 3.3.1.1.6

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be made consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.5 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required WRNM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

A Note is provided for Function 2.a that requires this SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

A second Note is provided for Function 2.b that clarifies that the CHANNEL FUNCTIONAL TEST for Function 2.b includes testing of the recirculation flow processing electronics, excluding the flow transmitters.

SR 3.3.1.1.13

This SR ensures that scrams initiated from the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP to ensure that the calibration is valid.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~26.7%~~ 26.3% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

REFERENCES (continued)

16. Deleted

17. Deleted

18. Deleted

19. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3 Single-Loop Operation," May 1980.

20. Setpoint Methodology for Peach Bottom Atomic Power Station and Limerick Generating Station, CC-MA-103-2001.

21. Backup Stability Protection (BSP) for Inoperable Option III Solutions, OG02-0119, July 17, 2002.

22. GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

23. GEH letter to NRC, "NEDC-33075P-A, Detect and Suppress Solution - Confirmation Density (DSS-CD) Analytical Limit (TAC No. MD0277)," October 29, 2008. (ADAMS Accession No. ML083040052).

24. 000N7936-R0, "Project Task Report – Exelon Generation Company LLC, Peach Bottom Atomic Power Station Unit 2 & 3 MELLLA+, Task T0202: Thermal-Hydraulic Stability," April 2014.

25. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high water level trip indirectly initiates a reactor scram from the main turbine trip (above ~~26.7%~~ 26.3% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires two DFCS channels per trip system of high water level trip instrumentation to be OPERABLE to ensure the feedwater pump turbines and main turbine will trip on a valid reactor vessel high water level signal. Two DFCS channels (one per trip system) are needed to provide trip signals in order for the feedwater and main turbine trips to occur.

Two level signals are also required to ensure a single sensor failure will not prevent the trips of the feedwater pump turbines and main turbine when reactor vessel water level is at the high water level reference point.

Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Trip setpoints are specified in the setpoint calculations. The trip setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setting less conservative than the trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic or design limits are derived from the limiting values of the process parameters obtained from the safety analysis or

(continued)

BASES

LCO (continued)

other appropriate documents. The Allowable Values are derived from the analytic or design limits, corrected for calibration, process, and instrument errors. A channel is inoperable if its actual trip setting is not within its required Allowable Value. The trip setpoints are determined from analytical or design limits, corrected for calibration, process and instrument errors, as well as, instrument drift. The trip setpoints determined in this manner provide adequate protection by assuring instrument and process uncertainties expected for the environment during the operating time for the associated channels are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ ~~23%~~ 22.6% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below ~~23%~~ 22.6% RTP; therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

(continued)

BASES

ACTIONS

B.1 (continued)

signal on a valid signal. This requires one channel per trip system to be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. Alternatively, the affected feedwater pump(s) and affected main turbine valve(s) may be removed from service since this performs the intended function of the instrumentation. As discussed in the Applicability section of the Bases, operation below ~~23%~~ 22.6% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP from full power conditions in an orderly manner and without challenging plant systems.

Required Action C.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable feedwater pump turbine or main turbine stop valve. The Note clarifies the situations under which the associated Required Action would be the appropriate Required Action.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations, consistent with the assumptions of the current plant specific setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the feedwater and main turbine stop valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a stop valve is incapable of operating, the associated instrumentation channels would be inoperable. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 14.5.2.2.

2. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

3. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

BACKGROUND (continued)

pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV—Closure and the TCV Fast Closure, Trip Oil Pressure—Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the neutron flux, heat flux, and pressurization transients, and to minimize the decrease in MCPR. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that utilize EOC-RPT, are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone so that the Safety Limit MCPR is not exceeded. Alternatively, APLHGR operating limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the LHGR operating limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)") for an inoperable EOC-RPT, as specified in the COLR, are sufficient to allow this LCO to be met. The EOC-RPT function is automatically disabled when turbine first stage pressure is < ~~26.7%~~ 26.3% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement.

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions, i.e., the TSV-Closure and the TCV Fast Closure, Trip Oil Pressure-Low Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Trip setpoints are specified in the plant design documentation. The trip setpoints are selected

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

<u>Turbine Stop Valve—Closure</u> (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are position switches associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV—Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV—Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV—Closure Allowable Value is selected to detect imminent TSV closure.

This EOC-RPT Function is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. Below ~~26.7%~~ 26.3% RTP, the Reactor Pressure—High and the Average Power Range Monitor (APRM) Scram Clamp Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

<u>Turbine Control Valve Fast Closure, Trip Oil Pressure - Low</u>

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure—Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases peak reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure—Low Function is such that two or more TCVs must be closed (pressure switch trips)

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

<u>Turbine Control Valve Fast Closure, Trip Oil Pressure—Low</u> (continued)

to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure—Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. Below ~~26.7%~~ 26.3% RTP, the Reactor Pressure—High and the APRM Scram Clamp Functions of the RPS are adequate to maintain the necessary safety margins.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.1 and 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a thermal limit violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < ~~26.7%~~ 26.3% RTP within 4 hours. Alternately, for an inoperable breaker (e.g., the breaker may be inoperable such that it will not open) the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~26.7%~~ 26.3% RTP from full power conditions in an orderly manner and without challenging plant systems.

Required Action C.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable RPT breaker. The Note clarifies the situations under which the associated Required Action would be the appropriate Required Action.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 5) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2.4

This SR ensures that an EOC-RPT initiated from the TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP to ensure that the calibration remains valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~26.7%~~ 26.3% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2.5

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criterion is included in Reference 6.

A Note to the Surveillance states that breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6. This is allowed since the time to open the contacts after energization of the trip coil and the arc suppression time are short and do not appreciably change, due to the design of the breaker opening device and the fact that the breaker is not routinely cycled.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2.5 (continued)

Response times cannot be determined at power because operation of final actuated devices is required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2.6

This SR ensures that the RPT breaker interruption time (arc suppression time plus time to open the contacts) is provided to the EOC-RPT SYSTEM RESPONSE TIME test. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Figure 7.9.4A, Sheet 3 of 3 (EOC-RPT logic diagram).

2. UFSAR, Section 7.9.4.4.3.

3. UFSAR, Section 14.5.1.2.4.

4. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved version.

5. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

6. Core Operating Limits Report.

7. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs during the depressurization transient in order to maintain reactor steam dome pressue > 700 psia. The MSIV closure results in a scram, thus reducing reactor power to < ~~23%~~ 22.6% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

6. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

7. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

8. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump flow and differential pressure patterns are established by plotting historical data as discussed in Reference 2.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds ~~23%~~ 22.6% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. UFSAR, Section 14.6.3.

2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980.

3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

4. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9 (continued)

pressure when the HPCI System diverts steam flow. Reactor steam pressure must be $\leq$ 1053 and $\geq$ ~~915~~ 910 psig to perform SR 3.5.1.8 and greater than or equal to the Electro-Hydraulic Control (EHC) System minimum pressure set with the EHC System controlling pressure (EHC System begins controlling pressure at a nominal 150 psig) and $\leq$ 175 psig to perform SR 3.5.1.9. Adequate steam flow is represented by at least 2 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable. Therefore, SR 3.5.1.8 and SR 3.5.1.9 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.10

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that either the HPCI System

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing an individual who can rapidly close the system vent flow path if directed.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be $\leq$ 1053 and $\geq$ ~~915~~ 910 psig to perform SR 3.5.3.3 and greater than or equal to the Electro-Hydraulic Control (EHC) System minimum pressure set with the EHC System controlling pressure (the EHC System begins controlling pressure at a nominal 150 psig) and $\leq$ 175 psig to perform SR 3.5.3.4. Alternately, auxiliary steam can be used to perform SR 3.5.3.4. Adequate steam flow is represented by at least 2 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. Alternately, the low pressure Surveillance test may be performed prior to startup using an auxiliary steam supply. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

(continued)

B 3.7 Plant SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is ~~22.4%~~ 21.96% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without safety relief valves opening or a reactor scram. The Main Turbine Bypass System consists of nine modulating type hydraulically actuated bypass valves mounted on a valve manifold. The manifold is connected with two steam lines to the four main steam lines upstream of the turbine stop valves. The bypass valves are controlled by the bypass control function of the Pressure Regulator and Turbine Generator Control System, as discussed in the UFSAR, Section 7.11.3 (Ref. 1). The bypass valves are normally closed. However, if the total steam flow signal exceeds the turbine control valve flow signal of the Pressure Regulator and Turbine Generator Control System, the bypass control function will output a bypass flow signal to the bypass valves. The bypass valves will then open sequentially to bypass the excess flow through connecting piping and a pressure reducing orifice to the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is expected to function during the electrical load rejection transient, the turbine trip transient, and the feedwater controller failure maximum demand transient, as described in the UFSAR, Section 14.5.1.1 (Ref. 2), Section 14.5.1.2.1 (Ref. 3), and Section 14.5.2.2 (Ref. 4). However, the feedwater controller maximum demand transient is the limiting licensing basis transient which defines the MCPR operating limit if the Main Turbine Bypass System is inoperable. Opening the bypass valves during the pressurization events mitigates the increase in reactor vessel pressure, which affects the MCPR during the event.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR operating limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the LHGR operating limits (LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”) may be applied to allow this LCO to be met. The operating limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the minimum number of bypass valves, specified in the COLR, to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analyses (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ ~~23%~~ 22.6% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the applicable safety analyses transients. As discussed in the Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < ~~23%~~ 22.6% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more required bypass valves as specified in the COLR inoperable), or the required thermal operating limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analyses may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the thermal operating limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the required thermal operating limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP. As discussed in the Applicability section, operation at < ~~23%~~ 22.6% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analyses. The response time limits are specified in COLR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 7.11.3.

2. UFSAR, Section 14.5.1.1.

3. UFSAR, Section 14.5.1.2.1.

4. UFSAR, Section 14.5.2.2.

5. Deleted

6. NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of ~~23%~~ RTP prevents any bundle from exceeding critical power and is a conservative limit when reactor pressure < 785 psig.

22.6%

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

LCO (continued)

With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative factor.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are assumed to occur at high power levels. Design calculations (Ref. 6) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the wide range neutron monitor period-short scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < ~~23%~~ 22.6% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. The

(continued)

BASES

ACTIONS

B.1 (continued)

allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ 22.6% RTP and then periodically thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ 22.6% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single Loop Operation," May 1980.

6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

7. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

8. Deleted

9. NEDO-30130-A, "Steady State Nuclear Methods," April 1985.

(continued)

BASES

REFERENCES (continued)

10. Deleted

11. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

12. Peach Bottom Unit 3 Core Operating Limits Report (COLR).

13. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 10) to analyze slow flow runout transients. The flow dependent operating limit, $MCPR_f$, is evaluated based on a single recirculation pump flow runout event (Ref. 9).

Power dependent MCPR limits ($MCPR_p$) are determined by the codes used to evaluate transients as described in Reference 2. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between ~~23%~~ 22.6% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below ~~23%~~ 22.6% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below ~~23%~~ 22.6% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at ~~23%~~ 22.6% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and

(continued)

BASES

APPLICABILITY (continued)

flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to ~~23%~~ 22.6% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the wide range neutron monitor period-short function provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < ~~23%~~ 22.6% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ 22.6% RTP and periodically thereafter. It is compared to the specified limits

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

in the COLR (Ref. 12) to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ 22.6% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4,"Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

3. UFSAR, Chapter 3.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 14.

6. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single Loop Operation," May 1980.

(continued)

BASES

REFERENCES (continued)

7. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

8. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

9. NEDC-32427P, "Peach Bottom Atomic Power Station Unit 3 Cycle 10 ARTS Thermal Limits Analyses," December 1994.

10. NEDO-30130-A, "Steady State Nuclear Methods," April 1985.

11. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

12. Peach Bottom Unit 3 Core Operating Limits Report (COLR).

13. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES (continued)

includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking.

Power-dependent and flow-dependent LHGR adjustment factors may also be provided per Reference 1 to ensure that fuel design limits are not exceeded due to the occurrence of a postulated transient event during operation at off-rated (less than 100%) reactor power or core flow conditions. These adjustment factors are applied, if required, per the COLR and decrease the allowable LHGR value.

Additionally, for single recirculation loop operation, an LHGR multiplier may be provided per Reference 1. This multiplier is applied per the COLR and decreases the allowable LHGR value. This additional margin may be necessary during SLO to account for the conservative analysis assumption of an earlier departure from nucleate boiling with only one recirculation loop available.

The LHGR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < ~~23%~~ 22.6% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ ~~23%~~ 22.6% RTP.

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The

(continued)

BASES

ACTIONS

A.1 (continued)

2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

22.6%

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < ~~23%~~ RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER TO < ~~23%~~ RTP in an orderly manner and without challenging plant systems.

(continued)

22.6%

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

22.6%

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~23%~~ RTP and periodically thereafter. It is compared to the specified limits in the COLR (Ref. 11) to ensure that the reactor is operating within the assumptions of the safety analysis. The 12 hour allowance after THERMAL POWER ≥ ~~23%~~ RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

22.6%

REFERENCES

1. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. UFSAR, Chapter 3.

3. UFSAR, Chapter 6.

4. UFSAR, Chapter 14.

5. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3, Single-Loop Operation," May 1980.

6. NEDC-32162P, "Maximum Extended Load Line Limit and ARTS Improvements Program Analyses for Peach Bottom Atomic Power Station Units 2 and 3," Revision 2, March 1995.

7. NEDC-33566P, "Safety Analysis Report for Exelon Peach Bottom Atomic Power Station Units 2 and 3, Constant Pressure Power Uprate," Revision 0.

8. NEDC-32163P, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," January 1993.

9. G-080-VC-400, "Peach Bottom Atomic Power Station Units 2 & 3 GNF2 ECCS-LOCA Evaluation," GE Hitachi Nuclear Energy, 0000-0100-8531-R1, March 2011.

10. NUREG-0800, Section 4.2, Subsection II.A.2(g), Revision 2, July 1981.

11. Peach Bottom Unit 3 Core Operating Limits Report (COLR).

12. G-080-VC-272, "Peach Bottom Atomic Power Station ECCS-LOCA Evaluation for GE14," General Electric Company, GENE-J11-03716-09-02P, July 2000.

13. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-High (Setdown) (continued)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux-High (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux-High (Setdown) Function will provide a secondary scram to the Wide Range Neutron Monitor Period-Short Function because of the relative setpoints. At higher power levels, it is possible that the Average Power Range Monitor Neutron Flux-High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux-High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed ~~23%~~ 22.6% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < ~~23%~~ 22.6% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < ~~23%~~ 22.6% RTP.

The Average Power Range Monitor Neutron Flux-High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux-High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power-High

The Average Power Range Monitor Simulated Thermal Power-High Function monitors average neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux-High Function Allowable Value. A note is included, applicable when the plant is in single recirculation loop operation per LCO 3.4.1, which requires the flow value, used in the Allowable Value equation, be reduced by ΔW. The value of ΔW

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Simulated Thermal Power-High (continued)

is established to conservatively bound the inaccuracy created in the core flow/drive flow correlation due to back flow in the jet pumps associated with the inactive recirculation loop. The Allowable Value thus maintains thermal margins essentially unchanged from those for two loop operation. The value of ΔW is plant specific and is defined in plant procedures. The Allowable Value equation for single loop operation is only valid for flows down to W = ΔW; the Allowable Value does not go below ~~61.5%~~ 60.3% RTP. This is acceptable because back flow in the inactive recirculation loop is only evident with drive flows of approximately 35% or greater (Reference 19). The Nominal Trip Setpoint (NTSP) and the as-found and as-left tolerances (Leave Alone Zone) were determined in accordance with Reference 10.

The Average Power Range Monitor Simulated Thermal Power-High Function is not specifically credited in the safety analysis but is intended to provide an additional margin of protection from transient induced fuel damage during operation where recirculation flow is reduced to below the minimum required for rated power operation. The Average Power Range Monitor Simulated Thermal Power-High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Neutron Flux-High Function will provide a scram signal before the Average Power Range Monitor Simulated Thermal Power-High Function setpoint is exceeded.

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function. The APRM flow processing logic is considered inoperable whenever it cannot deliver a flow signal less than or equal to actual Recirculation flow conditions for all steady state and transient reactor conditions while in Mode 1. Reduced or Downscale flow conditions due to planned maintenance or testing activities during derated plant conditions (i.e. end of cycle coast down) will result in conservative setpoints for the APRM Simulated Thermal Power-High function, thus maintaining that function operable.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

Reference 22 describes the Detect and Suppress-Confirmation Density (DSS-CD) long-term stability solution and the licensing basis Confirmation Density Algorithm (CDA). Reference 22 also describes the DSS-CD Armed Region and the three additional algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (PBDA), the amplitude based algorithm (ABA), and the growth rate algorithm (GRA). All four algorithms are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the CDA. The remaining three algorithms provide defense-in-depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY is based only on the CDA.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into cells for evaluation by the OPRM algorithms.

DSS-CD operability requires at least 8 responsive OPRM cells per channel. The DSS-CD software includes a self-check for the responsive OPRM cells; therefore, no SR is necessary.

The OPRM Upscale Function is required to be OPERABLE when the plant is ≥ ~~18%~~ 17.6% RTP, which is established as a power level that is greater than or equal to 5% below the lower boundary of the Armed Region. This requirement is designed to encompass the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The OPRM Upscale Function is automatically trip-enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ ~~23%~~ 22.6% RTP corresponding to the MCPR monitoring threshold and reactor recirculation drive flow, is less than 75% of rated flow. This region is the OPRM Armed Region. Note (h) allows for entry into the DSS-CD Armed Region without automatic arming of DSS-CD prior to completely passing through the DSS-CD Armed Region during the first startup and the first shutdown following DSS-CD implementation.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.f. Oscillation Power Range Monitor (OPRM) Upscale</u> (continued)

As described in Reference 22 and 24, the RTP values for the OPRM Upscale Function to be OPERABLE (≥ ~~18%~~ 17.6% RTP) and for the OPRM Upscale Function to be auto-enabled (≥ ~~23%~~ 22.6% RTP) are sufficiently conservative for protection of the plant against thermal-hydraulic instabilities. The basis for the 5% RTP difference between the OPRM Upscale OPERABLE (~~18%~~ 17.6% RTP) and OPRM Upscale auto-enable value (~~23%~~ 22.6% RTP) is to ensure that no credible event, e.g., loss of feed water heating, could result in a plant power excursion where an inoperable OPRM channel entered into the OPRM Armed Region. Peach Bottom plant specific analyses performed at these low power levels (Ref. 24) have demonstrated that any power excursion resulting from credible events is bounded by 5% RTP. In addition, both the core-wide and channel decay ratios at the OPRM Upscale auto-enabled values are extremely low as documented in Reference 22, which demonstrates the low possibility of thermal-hydraulic instabilities at low power and confirms the conservatisms in the OPRM Upscale Function auto-enable RTP value. The conservatisms in the determination of the values for OPRM Upscale Function OPERABLE and the OPRM Upscale Function auto-enabled sufficiently compensate for possible inaccuracy of the APRM simulated thermal power signal versus actual core thermal power at power levels < ~~23%~~ 22.6% RTP. Therefore, there is no need to perform any calibration of the APRM simulated thermal power signal to calculated power with RTP < ~~23%~~ 22.6% in order to determine the OPRM Upscale Function OPERABLE.

If any OPRM auto-enable setpoint is in a non-conservative condition, i.e., the OPRM Upscale is not auto-enabled with RTP ≥ ~~23%~~ 22.6% and reactor recirculation drive flow ≤ 75% of rated, the associated channel is considered inoperable for the OPRM Upscale Function. Alternatively, the auto-enable setpoint may be adjusted to place the channel in a conservative condition (armed). If placed in the armed condition, the channel is considered OPERABLE.

Note (h) reflects the need for plant data collection in order to test the DSS-CD equipment. Testing the DSS-CD equipment ensures its proper operation and prevents spurious reactor trips. Entry into the DSS-CD Armed Region without automatic arming of DSS-CD during this initial testing phase also allows for changes in plant operations to address maintenance or other operational needs. However, during this initial testing period, the OPRM Upscale Function is OPERABLE and DSS-CD operability and capability to automatically arm shall be maintained at recirculation drive flow rates above the DSS-CD Armed Region flow boundary.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Stop Valve—Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve—Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This Function is not required when THERMAL POWER is < ~~26.7%~~ 26.3% RTP since the Reactor Pressure—High and the Average Power Range Monitor Scram Clamp Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7 and the generator load rejection with bypass failure event. For these events, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

26.3%

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the relayed emergency trip supply oil pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

26.3%

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ RTP. This Function is not required when THERMAL POWER is < ~~26.7%~~ RTP, since the Reactor Pressure—High and the Average Power Range Monitor Scram Clamp Functions are adequate to maintain the necessary safety margins.

26.3%

10. Turbine Condenser—Low Vacuum

The Turbine Condenser—Low Vacuum Function protects the integrity of the main condenser by scramming the reactor and thereby decreasing the severity of the low condenser vacuum transient on the condenser. This function also ensures integrity of the reactor due to loss of its normal heat sink. The reactor scram on a Turbine Condenser—Low Vacuum signal will occur prior to a reactor scram from a Turbine Stop Valve—Closure signal. This function is not specifically credited in any accident analysis but is being retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

(continued)

BASES

ACTIONS (continued)

J.3

BSP is a temporary means for protection against thermal-hydraulic instability events. An extended period of inoperability without automatic trip capability is not justified. Consequently, the required channels are required to be restored to OPERABLE status within 120 days.

Based on engineering judgment, the likelihood of an instability event that could not be adequately handled by the use of the BSP Regions (See Action J.1) and the BSP Boundary (See Action J.2) during a 120-day period is negligibly small. The 120-day period is intended to allow for resolution of a variety of equipment problems (e.g., design changes, extensive analysis, or other unforeseen circumstances). This action is not intended to be used for operational convenience. Correction of most equipment failures or inoperabilities is expected to normally be accomplished within the completion times allowed for Actions for Conditions A and I.

A Note is provided to indicate that LCO 3.0.4 is not applicable. The intent of the note is to allow plant startup while operating within the 120-day Completion Time for Required Action J.3. The primary purpose of this exclusion is to allow an orderly completion of design and verification activities, in the event of a required design change, without undue impact on plant operation.

K.1

If the required channels are not restored to OPERABLE status and the Required Actions of J are not met within the associated Completion Times, then the plant must be placed in an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to less than 17.6% ~~18%~~ RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the specified operating power level from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.2 (continued)

A restriction to satisfying this SR when < ~~23%~~ 22.6% RTP is provided that requires the SR to be met only at ≥ ~~23%~~ 22.6% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < ~~23%~~ 22.6% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ ~~23%~~ 22.6% RTP, the Surveillance is required to have been satisfactorily performed in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above ~~23%~~ 22.6% if the Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding ~~23%~~ 22.6% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

(Not Used.)

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.1.5 and SR 3.3.1.1.6

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be made consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.5 is not required to be performed when entering MODE 2 from MODE 1, since testing of the MODE 2 required WRNM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

A Note is provided for Function 2.a that requires this SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2.

A second Note is provided for Function 2.b that clarifies that the CHANNEL FUNCTIONAL TEST for Function 2.b includes testing of the recirculation flow processing electronics, excluding the flow transmitters.

SR 3.3.1.1.13

This SR ensures that scrams initiated from the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP to ensure that the calibration is valid.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~26.7%~~ 26.3% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

REFERENCES (continued)

15. Deleted

16. Deleted

17. Deleted

18. Deleted

19. NEDO-24229-1, "Peach Bottom Atomic Power Station Units 2 and 3 Single-Loop Operation," May 1980.

20. Setpoint Methodology for Peach Bottom Atomic Power Station and Limerick Generating Station, CC-MA-103-2001.

21. Backup Stability Protection (BSP) for Inoperable Option III Solutions, OG02-0119, July 17, 2002.

22. GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect and Suppress Solution - Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

23. GEH letter to NRC, "NEDC-33075P-A, Detect and Suppress Solution - Confirmation Density (DSS-CD) Analytical Limit (TAC No. MD0277)," October 29, 2008. (ADAMS Accession No. ML083040052).

24. 000N7936-R0, "Project Task Report - Exelon Generation Company LLC, Peach Bottom Atomic Power Station Unit 2 & 3 MELLLA+, Task T0202: Thermal-Hydraulic Stability," April 2014.

25. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high water level trip indirectly initiates a reactor scram from the main turbine trip (above ~~26.7%~~ 26.3% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires two DFCS channels per trip system of high water level trip instrumentation to be OPERABLE to ensure the feedwater pump turbines and main turbine will trip on a valid reactor vessel high water level signal. Two DFCS channels (one per trip system) are needed to provide trip signals in order for the feedwater and main turbine trips to occur.

Two level signals are also required to ensure a single sensor failure will not prevent the trips of the feedwater pump turbines and main turbine when reactor vessel water level is at the high water level reference point.

Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Trip setpoints are specified in the setpoint calculations. The trip setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setting less conservative than the trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic or design limits are derived from the limiting values of the process parameters obtained from the safety analysis or

(continued)

BASES

LCO (continued)	other appropriate documents. The Allowable Values are derived from the analytic or design limits, corrected for calibration, process, and instrument errors. A channel is inoperable if its actual trip setting is not within its required Allowable Value. The trip setpoints are determined from analytical or design limits, corrected for calibration, process and instrument errors, as well as, instrument drift. The trip setpoints determined in this manner provide adequate protection by assuring instrument and process uncertainties expected for the environment during the operating time for the associated channels are accounted for.
APPLICABILITY	The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ ~~23%~~ 22.6% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 22.6% ~~23%~~ RTP; therefore, these requirements are only necessary when operating at or above this power level.
ACTIONS	A Note has been provided to modify the ACTIONS related to feedwater and main turbine high water level trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable feedwater and main turbine high water level trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable feedwater and main turbine high water level trip instrumentation channel.

(continued)

BASES

ACTIONS

B.1 (continued)

signal on a valid signal. This requires one channel per trip system to be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP within 4 hours. Alternatively, the affected feedwater pump(s) and affected main turbine valve(s) may be removed from service since this performs the intended function of the instrumentation. As discussed in the Applicability section of the Bases, operation below ~~23%~~ 22.6% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < ~~23%~~ 22.6% RTP from full power conditions in an orderly manner and without challenging plant systems.

Required Action C.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable feedwater pump turbine or main turbine stop valve. The Note clarifies the situations under which the associated Required Action would be the appropriate Required Action.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations, consistent with the assumptions of the current plant specific setpoint methodology.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.2.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the feedwater and main turbine stop valves is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a stop valve is incapable of operating, the associated instrumentation channels would be inoperable. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 14.5.2.2.

2. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

3. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

BACKGROUND (continued)

pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV—Closure and the TCV Fast Closure, Trip Oil Pressure—Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the neutron flux, heat flux, and pressurization transients, and to minimize the decrease in MCPR. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that utilize EOC-RPT, are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone so that the Safety Limit MCPR is not exceeded. Alternatively, APLHGR operating limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the LHGR operating limits (LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”) for an inoperable EOC-RPT, as specified in the COLR, are sufficient to allow this LCO to be met. The EOC-RPT function is automatically disabled when turbine first stage pressure is < ~~26.7%~~ 26.3% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement.

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions, i.e., the TSV-Closure and the TCV Fast Closure, Trip Oil Pressure-Low Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Trip setpoints are specified in the plant design documentation. The trip setpoints are selected

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

Turbine Stop Valve—Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are position switches associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV—Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV—Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV—Closure Allowable Value is selected to detect imminent TSV closure.

This EOC-RPT Function is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. Below ~~26.7%~~ 26.3% RTP, the Reactor Pressure—High and the Average Power Range Monitor (APRM) Scram Clamp Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure—Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases peak reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure—Low Function is such that two or more TCVs must be closed (pressure switch trips)

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP as measured at the turbine first stage pressure. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure—Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. Below ~~26.7%~~ 26.3% RTP, the Reactor Pressure—High and the APRM Scram Clamp Functions of the RPS are adequate to maintain the necessary safety margins.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.1 and 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a thermal limit violation.

C.1 and C.2

26.3%

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < ~~26.7%~~ RTP within 4 hours. Alternately, for an inoperable breaker (e.g., the breaker may be inoperable such that it will not open) the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~26.7%~~ 26.3% RTP from full power conditions in an orderly manner and without challenging plant systems.

Required Action C.1 is modified by a Note which states that the Required Action is only applicable if the inoperable channel is the result of an inoperable RPT breaker. The NOTE clarifies the situations under which the associated Required Action would be the appropriate Required Action.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 5) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2.4

This SR ensures that an EOC-RPT initiated from the TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~26.7%~~ 26.3% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ ~~26.7%~~ 26.3% RTP to ensure that the calibration remains valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~26.7%~~ 26.3% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2.5

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criterion is included in Reference 6.

A Note to the Surveillance states that breaker interruption time may be assumed from the most recent performance of SR 3.3.4.2.6. This is allowed since the time to open the contacts after energization of the trip coil and the arc suppression time are short and do not appreciably change, due to the design of the breaker opening device and the fact that the breaker is not routinely cycled.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2.5 (continued)

Response times cannot be determined at power because operation of final actuated devices is required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.4.2.6

This SR ensures that the RPT breaker interruption time (arc suppression time plus time to open the contacts) is provided to the EOC-RPT SYSTEM RESPONSE TIME test. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Figure 7.9.4A, Sheet 3 of 3 (EOC-RPT logic diagram).

2. UFSAR, Section 7.9.4.4.3.

3. UFSAR, Section 14.5.1.2.4.

4. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved version.

5. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

6. Core Operating Limits Report.

7. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < ~~23%~~ 22.6% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.7

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 7.3.

2. NRC Safety Evaluation Report for Amendment Numbers 156 and 158 to Facility Operating License Numbers DPR-44 and DPR-56, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, September 7, 1990.

3. UFSAR, Chapter 14.

4. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

5. UFSAR, Section 4.9.3.

6. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

7. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

8. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump flow and differential pressure patterns are established by plotting historical data as discussed in Reference 2.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

22.6%

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds ~~23%~~ of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. UFSAR, Section 14.6.3.

2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980.

3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

4. NEDC-33873P, "Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9 (continued)

910

pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≤ 1053 and ≥ ~~915~~ psig to perform SR 3.5.1.8 and greater than or equal to the Electro-Hydraulic Control (EHC) System minimum pressure set with the EHC System controlling pressure (EHC System begins controlling pressure at a nominal 150 psig) and ≤ 175 psig to perform SR 3.5.1.9. Adequate steam flow is represented by at least 2 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that HPCI is inoperable. Therefore, SR 3.5.1.8 and SR 3.5.1.9 are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.1.10

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCI, CS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. This SR also ensures that either the HPCI System

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by a Note which exempts system vent flow paths opened under administrative control. The administrative control should be proceduralized and include stationing an individual who can rapidly close the system vent flow path if directed.

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow is tested both at the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≤ 1053 and ≥ ~~915~~ 910 psig to perform SR 3.5.3.3 and greater than or equal to the Electro-Hydraulic Control (EHC) System minimum pressure set with the EHC System controlling pressure (the EHC System begins controlling pressure at a nominal 150 psig) and ≤ 175 psig to perform SR 3.5.3.4. Alternately, auxiliary steam can be used to perform SR 3.5.3.4. Adequate steam flow is represented by at least 2 turbine bypass valves open. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance is short. Alternately, the low pressure Surveillance test may be performed prior to startup using an auxiliary steam supply. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure Surveillance has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test.

(continued)

B 3.7 Plant SYSTEMS

B 3.7.6 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is ~~22.4%~~ 21.96% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without safety relief valves opening or a reactor scram. The Main Turbine Bypass System consists of nine modulating type hydraulically actuated bypass valves mounted on a valve manifold. The manifold is connected with two steam lines to the four main steam lines upstream of the turbine stop valves. The bypass valves are controlled by the bypass control unit of the Pressure Regulator and Turbine Generator Control System, as discussed in the UFSAR, Section 7.11.3 (Ref. 1). The bypass valves are normally closed. However, if the total steam flow signal exceeds the turbine control valve flow signal of the Pressure Regulator and Turbine Generator Control System, the bypass control unit processes these signals and will output a bypass flow signal to the bypass valves. The bypass valves will then open sequentially to bypass the excess flow through connecting piping and a pressure reducing orifice to the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is expected to function during the electrical load rejection transient, the turbine trip transient, and the feedwater controller failure maximum demand transient, as described in the UFSAR, Section 14.5.1.1 (Ref. 2), Section 14.5.1.2.1 (Ref. 3), and Section 14.5.2.2 (Ref. 4). However, the feedwater controller maximum demand transient is the limiting licensing basis transient which defines the MCPR operating limit if the Main Turbine Bypass System is inoperable. Opening the bypass valves during the pressurization events mitigates the increase in reactor vessel pressure, which affects the MCPR during the event.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR operating limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), the MCPR operating limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and the LHGR operating limits (LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”) may be applied to allow this LCO to be met. The operating limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the minimum number of bypass valves, specified in the COLR, to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analyses (Refs. 2, 3, and 4).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ ~~23%~~ 22.6% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the applicable safety analyses transients. As discussed in the Bases for LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < ~~23%~~ 22.6% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more required bypass valves as specified in the COLR inoperable), or the required thermal operating limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analyses may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the thermal operating limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the required thermal operating limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < ~~23%~~ 22.6% RTP. As discussed in the Applicability section, operation at < ~~23%~~ 22.6% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable safety analyses transients. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.6.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analyses. The response time limits are specified in COLR. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 7.11.3.

2. UFSAR, Section 14.5.1.1.

3. UFSAR, Section 14.5.1.2.1.

4. UFSAR, Section 14.5.2.2.

5. Deleted

6. NEDC-33873P, “Safety Analysis for Peach Bottom Atomic Power Station, Units 2 and 3, Thermal Power Optimization," Revision 0.

TRMS 1.1 Definitions (continued)

OPERATIONS WITH THE POTENTIAL FOR DRAINING THE REACTOR VESSEL (OPDRVs)

Plain language meaning:

ANY activity that could potentially result in draining or siphoning the RPV water level below the top of the fuel, without taking credit for mitigating measures, would be an OPDRV activity.

--Note--

On 10/4/11, the NRC issued an Enforcement Guidance Memorandum (EGM) involving plants that do not use the above 'plain language' meaning of OPDRV as intended by the NRC. Until 12/31/13, the NRC will provide enforcement discretion for non-compliances with use of the plain language meaning of OPDRVs as long as all the conditions of the EGM are satisfied. Therefore, approved procedure(s) are required to be in place to fully implement all the conditions specified in the NRC EGM if not in full compliance with the plain language meaning. (Reference IR 1273127)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3951~~ 4016 MWt.

Table 3.2-1 (Page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	TEST REQUIREMENTS	ALLOWABLE VALUE
3. APRM				
a. Simulated Thermal Power-High	1	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	<u>Two Loop Operation:</u> $\leq$ ~~0.61 W_d + 57.5%~~ 0.60 W_d + 56.5% (Clamp @ 108.4% max) <u>Single Loop Operation:</u> $\leq$ ~~0.55 (W_d - ΔW) + 51.9%~~ 0.54 (W_d - ΔW) + 50.9% (Clamp @ 108.4% max)
b. Simulated Thermal Power-High (Setdown)	$2^{(a)}$	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	$\leq$ 12%
c. Downscale	1	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	$\geq$ 2.8 indicated on scale
4. Scram Discharge Instrument Volume High Level	1, $2^{(a)}$, $5^{(g)}$	1	TR 3.2.4 TR 3.2.7 TR 3.2.8	$\leq$ 25 gallons

(a) With mode switch in startup.

(g) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

B 3.2 CONTROL ROD BLOCK INSTRUMENTATION

BASES

The objectives of the Control Rod Block Instrumentation TRMS are to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out-of-service for maintenance and to prescribe the trip settings required to assure adequate performance. The trip settings are chosen at a level away from the normal operating range to prevent inadvertent actuation of the control rod block instrumentation involved and exposure to abnormal situations.

The trip logic for the control rod block functions is 1 out of n: e.g., any trip on one of 4 APRMs or 8 WRNMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met.

The Average Power Range Monitor (APRM) rod block Function in MODE 1 (Function 3.a) is flow biased and based on Simulated Thermal Power (STP); it prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection: i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The APRM control rod block trip setting (Function 3.a) shall be as follows:

For Two Loop Operation: $\boxed{0.60\ W_d + 56.5\%}$

$\le$ ~~$0.61\ W_d + 57.5\%$~~ (Clamp @ 108.4%)

For Single Loop Operation: $\boxed{0.54\ (W_d - \Delta W) + 50.9\%}$

$\le$ ~~$0.55\ (W_d - \Delta W) + 51.9\%$~~ (Clamp @ 108.4%)

Where: $\boxed{4016}$

S_{RB} = Control rod block setting in percent of rated thermal power (~~3951~~ MWt)

W_d = Loop recirculation flow rate in percent of design.

ΔW = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single $\boxed{-0.54}$ loop operation, the reduction in trip setting (~~-0.55~~ ΔW) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM rod block setpoint and recirculation drive flow or by adjusting the APRM rod block trip setting. $\Delta W = 0$ for two loop operation.

The APRM control rod block trip setting shall not exceed 108.4% of rated thermal power.

(continued)

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.6.1	Perform CHANNEL CHECK.	12 hours
TR 3.6.2	---------------------NOTE--------------------------- Not required to be performed until 12 hours After thermal power ≥ ~~23%~~ 22.6% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~23%~~ 22.6% RTP.	7 days
TR 3.6.3	Perform CHANNEL CHECK.	31 days
TR 3.6.4	Perform CHANNEL CALIBRATION.	184 days
TR 3.6.5	Perform CHANNEL CALIBRATION.	18 months
TR 3.6.6	Perform CHANNEL FUNCTIONAL TEST.	24 months
TR 3.6.7	Perform CHANNEL CALIBRATION.	24 months
TR 3.6.8	Perform CHANNEL CALIBRATION.	92 days

3.20 LEADING EDGE FLOW METER (LEFM) SYSTEM

TRMS 3.20 Three LEFM flow meters shall be NORMAL

APPLICABILITY: MODE 1 with THERMAL POWER > 3951 MWt

COMPENSATORY MEASURES

-- NOTES --

See Bases for Definitions of LEFM flow meter NORMAL, MAINTENANCE, and FAIL status.

Separate Condition entry is allowed for each LEFM flow meter.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One or more LEFM flow meter in MAINTENANCE	A.1 Restore three LEFM flow meters to NORMAL, ensuring all flow input to the Core Thermal Power Calculation is from LEFM flow meters.	72 hours
B. Required Action and associated Completion Time of Condition A not met	B.1 Reduce Thermal Power to less than or equal to 4010 MWt.	Immediately
	<u>AND</u>	
	B.2 Ensure all flow input to the Core Thermal Power Calculation is from LEFM flow meters.	Immediately

C. One or More LEFM flow meter in FAIL	C.1 Reduce Thermal Power to less than or equal to 3951 MWt.	72 hours

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.20.1	Perform CHANNEL CALIBRATION	24 months

B 3.20 LEADING EDGE FLOW METER (LEFM) SYSTEM

BASES

This TRMS is provided to ensure that Core Thermal Power (CTP) is maintained at a level consistent with the feedwater flow measurement uncertainty. The three LEFM Flow Meters shall be NORMAL for power operations above 3951 MWt OR CTP must be limited in accordance with this TRMS. This TRMS allows Separate Condition Entry for each LEFM Flow Meter.

The LEFM System consists of three LEFM Flow Meters, one in each of the three feedwater lines. Each Flow Meter contains flow transducers arranged in two planes. Plane 1 consists of flow transducer paths 1 through 4 and Plane 2 consists of flow transducer paths 5 through 8. The flow data from a LEFM flow meter with a single functioning plane has greater associated measurement uncertainty than that from a LEFM flow meter with both planes functioning, but less associated measurement uncertainty than that from a feedwater flow nozzle (Venturi).

The LEFM System computer converts the LEFM Flow Meter data into feedwater flow and temperature signals for that loop and provides a self-check and flow measurement uncertainty determination (via Plant Monitoring System alarms). There are three possible statuses for a LEFM flow meter; NORMAL, MAINTENANCE, and FAIL.

A LEFM Flow Meter status is considered NORMAL IF:

The LEFM System Computer indicates that flow meter status (mode) to be Normal.

A LEFM flow meter status is considered MAINTENANCE IF:

The LEFM System Computer indicates that flow meter status (mode) to be Maintenance.

A LEFM flow meter status is considered FAIL IF:

The LEFM System Computer indicates that flow meter status (mode) to be Fail.

The LEFM Flow Meter status is determined and reported by the LEFM System computer based upon the number of functional Planes in the LEFM Flow Meter and upon its data quality. For additional background information on the criteria used by the LEFM System computer to determine the status of an individual LEFM Flow Meter, see References 1 & 2.

When this TRMS is applicable (>3951 MWt) and except as explicitly directed otherwise in the TRM, the feedwater flow input to the Core Thermal Power calculation from a LEFM Flow Meter that is not NORMAL is to be replaced with that from the associated calibrated feedwater flow nozzle (Venturi). A feedwater flow nozzle is calibrated when a correction factor based on the LEFM/Venturi ratio is applied to the feedwater flow nozzle measurement in accordance with station operating procedures. This will ensure accuracy of the core thermal power calculation while relying on the feedwater flow nozzle input to the Core Thermal Power calculation. See Reference 3.

The feedwater flow signal from an LEFM Flow Meter in FAIL status is to remain replaced by its corresponding feedwater flow nozzle as long as the LEFM Flow Meter remains in FAIL.

The feedwater flowrate signal from a LEFM Flow Meter in MAINTENANCE status is to provide input to the Core Thermal Power calculation when operating at the intermediate power level ≤ 4010 MWt in accordance with Required Compensatory Measure B.1. This intermediate power level is predicated upon all three feedwater flow inputs being provided by LEFM Flow Meters that are all in either NORMAL or MAINTENANCE status.

If all three LEFM Flow Meters are restored to the NORMAL status after entry into Required Compensatory Measure B.1, then all three LEFM Flow Meters must provide feedwater flow input to the Core Thermal Power calculation prior to raising power >4010 MWt.

If the status of an LEFM Flow Meter changes to a status other than NORMAL after a TRM Condition has been entered for that Flow Meter (i.e. status from MAINTENANCE to FAIL or FAIL to MAINTENANCE), then the Completion Time(s) for the new Required Compensatory Measure(s) of the applicable TRM Condition(s) must be completed based upon a start time corresponding to initial entry into the TRM for the specific LEFM Flow Meter. The accuracy of the calibrated feedwater flow nozzle (Venturi) can only be credited for 72 hours based on the insignificant instrument drift, see Reference 3. If the LEFM Flow Meter cannot be restored to NORMAL in the 72 hour Completion Time, then CTP must be lowered as directed by the appropriate Required Compensatory Measure based on LEFM Flow Meter status at the time.

The analysis supporting the allowable power levels provided in the TRM is contained in References 1 & 2.

The LEFM system calibration will be checked at regularly scheduled intervals. The frequency has been selected based on the reliability of the system. The calibration includes appropriate heat balance parameters required for LEFM System operation above 3951 MWt.

REFERENCES:

1. Calculation PM-1201, Uncertainty Analysis for Thermal Power Determination at PB2 Using the LEFM CheckPlus System, VNDR DWG NUMBER ER464
2. Calculation PM-1202, Uncertainty Analysis for Thermal Power Determination at PB3 Using the LEFM CheckPlus System, VNDR DWG NUMBER ER463
3. Technical Evaluation 2677307-06, LEFM Basis Information for TPO Uprate LAR

TRMS 1.1 Definitions (continued)

OPERATIONS WITH THE POTENTIAL FOR DRAINING THE REACTOR VESSEL (OPDRVs)

Plain language meaning:

ANY activity that could potentially result in draining or siphoning the RPV water level below the top of the fuel, without taking credit for mitigating measures, would be an OPDRV activity.

---Note---
On 10/4/11, the NRC issued an Enforcement Guidance Memorandum (EGM) involving plants that do not use the above 'plain language' meaning of OPDRV as intended by the NRC. Until 12/31/13, the NRC will provide enforcement discretion for non-compliances with use of the plain language meaning of OPDRVs as long as all the conditions of the EGM are satisfied. Therefore, approved procedure(s) are required to be in place to fully implement all the conditions specified in the NRC EGM if not in full compliance with the plain language meaning. (Reference IR 1273127)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3951~~ 4016 MWt.

Table 3.2-1 (Page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	TEST REQUIREMENTS	ALLOWABLE VALUE
3. APRM				
a. Simulated Thermal Power-High	1	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	Two Loop Operation: $\leq$ $\cancel{0.61\,W_d + 57.5\%}$ $0.60W_d + 56.5\%$ (Clamp @ 108.4% max) Single Loop Operation: $\leq$ $\cancel{0.55\,(W_d - \Delta W) + 51.9\%}$ $0.54\,(W_d - \Delta W) + 50.9\%$ (Clamp @ 108.4% max)
b. Simulated Thermal Power-High (Setdown)	2[a]	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	$\leq$ 12%
c. Downscale	1	3	TR 3.2.2 TR 3.2.5 TR 3.2.8	$\geq$ 2.8 indicated on scale
4. Scram Discharge Instrument Volume High Level	1, 2[a], 5[g]	1	TR 3.2.4 TR 3.2.7 TR 3.2.8	$\leq$ 25 gallons

(a) With mode switch in startup.

(g) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

B 3.2 CONTROL ROD BLOCK INSTRUMENTATION

BASES

The objectives of the Control Rod Block Instrumentation TRMS are to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out-of-service for maintenance and to prescribe the trip settings required to assure adequate performance. The trip settings are chosen at a level away from the normal operating range to prevent inadvertent actuation of the control rod block instrumentation involved and exposure to abnormal situations.

The trip logic for the control rod block functions is 1 out of n: e.g., any trip on one of 4 APRMs or 8 WRNMs will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met.

The Average Power Range Monitor (APRM) rod block Function in MODE 1 (Function 3.a) is flow biased and based on Simulated Thermal Power (STP); it prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection: i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The APRM control rod block trip setting (Function 3.a) shall be as follows:

<u>For Two Loop Operation:</u>
≤ ~~0.61 W_d + 57.5%~~ 0.60 W_d + 56.5% (Clamp @ 108.4%)

<u>For Single Loop Operation:</u>
≤ ~~0.55 (W_d - ΔW) + 51.9%~~ 0.54 (W_d - ΔW) + 50.9% (Clamp @ 108.4%)

Where:

S_{RB} = Control rod block setting in percent of rated thermal power (~~3951~~ 4016 MWt)

W_d = Loop recirculation flow rate in percent of design.

ΔW = Difference between two loop and single loop effective recirculation drive flow at the same core flow. During single loop operation, the reduction in trip setting (~~-0.55~~ -0.54 ΔW) is accomplished by correcting the flow input of the flow biased rod block to preserve the original (two loop) relationship between APRM rod block setpoint and recirculation drive flow or by adjusting the APRM rod block trip setting. ΔW = 0 for two loop operation.

The APRM control rod block trip setting shall not exceed 108.4% of rated thermal power.

(continued)

TEST REQUIREMENTS

	TEST	FREQUENCY
TR 3.6.1	Perform CHANNEL CHECK.	12 hours
TR 3.6.2	---------------------NOTE-------------------------- Not required to be performed until 12 hours After thermal power ≥ ~~23%~~ 22.6% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ ~~23%~~ 22.6% RTP.	7 days
TR 3.6.3	Perform CHANNEL CHECK.	31 days
TR 3.6.4	Perform CHANNEL CALIBRATION.	184 days
TR 3.6.5	Perform CHANNEL CALIBRATION.	18 months
TR 3.6.6	Perform CHANNEL FUNCTIONAL TEST.	24 months
TR 3.6.7	Perform CHANNEL CALIBRATION.	24 months
TR 3.6.8	Perform CHANNEL CALIBRATION.	92 days

3.20 LEADING EDGE FLOW METER (LEFM) SYSTEM

TRMS 3.20 Three LEFM flow meters shall be NORMAL

APPLICABILITY: MODE 1 with THERMAL POWER > 3951 MWt

COMPENSATORY MEASURES

-- NOTES --

See Bases for Definitions of LEFM flow meter NORMAL, MAINTENANCE, and FAIL status.

Separate Condition entry is allowed for each LEFM flow meter.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One or more LEFM flow meter in MAINTENANCE	A.1 Restore three LEFM flow meters to NORMAL, ensuring all flow input to the Core Thermal Power Calculation is from LEFM flow meters.	72 hours
B. Required Action and associated Completion Time of Condition A not met	B.1 Reduce Thermal Power to less than or equal to 4010 MWt.	Immediately
	<u>AND</u>	
	B.2 Ensure all flow input to the Core Thermal Power Calculation is from LEFM flow meters.	Immediately

C. One or More LEFM flow meter in FAIL	C.1 Reduce Thermal Power to less than or equal to 3951 MWt.	72 hours

TEST REQUIREMENTS

TEST		FREQUENCY
TR 3.20.1	Perform CHANNEL CALIBRATION	24 months

B 3.20 LEADING EDGE FLOW METER (LEFM) SYSTEM

BASES

This TRMS is provided to ensure that Core Thermal Power (CTP) is maintained at a level consistent with the feedwater flow measurement uncertainty. The three LEFM Flow Meters shall be NORMAL for power operations above 3951 MWt OR CTP must be limited in accordance with this TRMS. This TRMS allows Separate Condition Entry for each LEFM Flow Meter.

The LEFM System consists of three LEFM Flow Meters, one in each of the three feedwater lines. Each Flow Meter contains flow transducers arranged in two planes. Plane 1 consists of flow transducer paths 1 through 4 and Plane 2 consists of flow transducer paths 5 through 8. The flow data from a LEFM flow meter with a single functioning plane has greater associated measurement uncertainty than that from a LEFM flow meter with both planes functioning, but less associated measurement uncertainty than that from a feedwater flow nozzle (Venturi).

The LEFM System computer converts the LEFM Flow Meter data into feedwater flow and temperature signals for that loop and provides a self-check and flow measurement uncertainty determination (via Plant Monitoring System alarms). There are three possible statuses for a LEFM flow meter; NORMAL, MAINTENANCE, and FAIL.

A LEFM Flow Meter status is considered NORMAL IF:

> The LEFM System Computer indicates that flow meter status (mode) to be Normal.

A LEFM flow meter status is considered MAINTENANCE IF:

> The LEFM System Computer indicates that flow meter status (mode) to be Maintenance.

A LEFM flow meter status is considered FAIL IF:

> The LEFM System Computer indicates that flow meter status (mode) to be Fail.

The LEFM Flow Meter status is determined and reported by the LEFM System computer based upon the number of functional Planes in the LEFM Flow Meter and upon its data quality. For additional background information on the criteria used by the LEFM System computer to determine the status of an individual LEFM Flow Meter, see References 1 & 2.

When this TRMS is applicable (>3951 MWt) and except as explicitly directed otherwise in the TRM, the feedwater flow input to the Core Thermal Power calculation from a LEFM Flow Meter that is not NORMAL is to be replaced with that from the associated calibrated feedwater flow nozzle (Venturi). A feedwater flow nozzle is calibrated when a correction factor based on the LEFM/Venturi ratio is applied to the feedwater flow nozzle measurement in accordance with station operating procedures. This will ensure accuracy of the core thermal power calculation while relying on the feedwater flow nozzle input to the Core Thermal Power calculation. See Reference 3.

The feedwater flow signal from an LEFM Flow Meter in FAIL status is to remain replaced by its corresponding feedwater flow nozzle as long as the LEFM Flow Meter remains in FAIL.

The feedwater flowrate signal from a LEFM Flow Meter in MAINTENANCE status is to provide input to the Core Thermal Power calculation when operating at the intermediate power level ≤ 4010 MWt in accordance with Required Compensatory Measure B.1. This intermediate power level is predicated upon all three feedwater flow inputs being provided by LEFM Flow Meters that are all in either NORMAL or MAINTENANCE status.

If all three LEFM Flow Meters are restored to the NORMAL status after entry into Required Compensatory Measure B.1, then all three LEFM Flow Meters must provide feedwater flow input to the Core Thermal Power calculation prior to raising power >4010 MWt.

If the status of an LEFM Flow Meter changes to a status other than NORMAL after a TRM Condition has been entered for that Flow Meter (i.e. status from MAINTENANCE to FAIL or FAIL to MAINTENANCE), then the Completion Time(s) for the new Required Compensatory Measure(s) of the applicable TRM Condition(s) must be completed based upon a start time corresponding to initial entry into the TRM for the specific LEFM Flow Meter. The accuracy of the calibrated feedwater flow nozzle (Venturi) can only be credited for 72 hours based on the insignificant instrument drift, see Reference 3. If the LEFM Flow Meter cannot be restored to NORMAL in the 72 hour Completion Time, then CTP must be lowered as directed by the appropriate Required Compensatory Measure based on LEFM Flow Meter status at the time.

The analysis supporting the allowable power levels provided in the TRM is contained in References 1 & 2.

The LEFM system calibration will be checked at regularly scheduled intervals. The frequency has been selected based on the reliability of the system. The calibration includes appropriate heat balance parameters required for LEFM System operation above 3951 MWt.

REFERENCES:
1. Calculation PM-1201, Uncertainty Analysis for Thermal Power Determination at PB2 Using the LEFM CheckPlus System, VNDR DWG NUMBER ER464
2. Calculation PM-1202, Uncertainty Analysis for Thermal Power Determination at PB3 Using the LEFM CheckPlus System, VNDR DWG NUMBER ER463
3. Technical Evaluation 2677307-06, LEFM Basis Information for TPO Uprate LAR

ATTACHMENT 4

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

NRC Regulatory Issue Summary 2002-03 Cross-Reference

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty			
I.1.	Detailed description of plant-specific implementation of feedwater flow measurement technique and power increase gained as a result of implementing technique	Attachment 1	3.2 General Approach 3.3 LEFM Ultrasonic Flow Measurement and Core Thermal Power Uncertainty
I.1.A.	NRC approval of topical report on flow measurement technique	Attachment 1 Attachment 5	3.3.1 LEFM Flow Measurement 1.1 Overview
I.1.B.	Reference to NRC's approval of proposed measurement technique	Attachment 1 Attachment 5	3.3.1 LEFM Flow Measurement 1.1 Overview
I.1.C.	Plant Implementation	Attachment 1	3.3.2 Plant Implementation
I.1.D.	Disposition of NRC criteria	Attachment 1	3.2.4 Disposition of NRC Criteria for Use of LEFM Topical Reports
I.1.E.	Total power measurement uncertainty calculation for the plant	Attachment 1 Attachment 8	3.3.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology Cameron ER-464 "Bounding Uncertainty Analysis for Thermal Power Determination at PBAPS Unit 2 Using the LEFM CheckPlus System," (Proprietary Version), and ER-463, "Bounding Uncertainty Analysis for Thermal Power Determination at PBAPS Unit 3 Using the LEFM CheckPlus System," (Proprietary Version)
I.1.F.	Calibration and maintenance	Attachment 1	3.3.4 Disposition of NRC Criteria for Use of Topical Reports 3.3.5 Deficiencies and Corrective Actions
I.1.G.	Proposed outage time for LEFM and basis for selected time	Attachment 1 Attachment 3	3.3.4 Disposition of NRC Criteria for Use of Topical Reports TRM 3.20 Leading Edge Flow Meter (LEFM) System

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
I.1.H	Proposed actions if outage time is exceeded, and basis for actions	Attachment 1 Attachment 3	3.3.4 Disposition of NRC Criteria for Use of Topical Reports TRM 3.20 Leading Edge Flow Meter (LEFM) System
II. Accidents and Transients for which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level			
II.1	Matrix for bounded accidents and transients	Attachment 5	9.0 Reactor Safety Performance Evaluations
III. Accidents and Transients for which the Existing Analyses of Record Do Not Bound Plant Operation At the Proposed Uprated Power Level			
III.1,2, 3	Matrix for unbounded accidents and transients	Attachment 5	9.0 Reactor Safety Performance Evaluations
IV. Mechanical/Structural/Material Component Integrity and Design			
IV.1.A.i	Reactor vessel, nozzles, supports	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness 3.2.2 Reactor Vessel Structural Evaluation
IV.1.A.ii	Reactor core support structures and vessel internals	Attachment 5 Attachment 1 Attachment 10	3.3 Reactor Internals 3.3.1 Reactor Internal Pressure Difference 3.3.2 Reactor Internals Structural Evaluation 3.3.3 Steam Separator and Dryer Performance 3.4 Flow-Induced Vibration 3.5.2 Adverse Flow Effects Westinghouse Electric Company (WEC), Peach Bottom Units 2 and 3 Steam Dryer Structural Analysis Results at MUR Conditions, dated October 27, 2016

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
IV.1.A.iii	Control rod drive mechanisms	Attachment 5	2.5 Reactivity Control
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	Attachment 5	3.4 Flow-Induced Vibration 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.6 Reactor Recirculation System 3.7 Main Steam Line Flow Restrictors 3.8 Main Steam Isolation Valves 3.9 Reactor Core Isolation Cooling 3.10 Residual Heat Removal System 3.11 Reactor Water Cleanup System
IV.1.A.v	Balance of plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, and containment systems)	Attachment 5	3.5 Piping Evaluation 3.5.2 Balance-of-Plant Piping Evaluation 6.4.1 Service Water Systems 6.4.3 Chilled Water System 6.4.5 Reactor Building Closed Cooling Water System
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	NA	NA
IV.1.A.vii	Reactor coolant pumps	NA	NA
IV.1A.viii	Pressurizer shell, nozzles, and surge lines	NA	NA

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
IV.1.A.ix	Safety-related valves	Attachment 5	3.1 Nuclear System Pressure Relief/Overpressure Protection 3.8 Main Steam Isolation Valves 4.1 Containment System Performance 4.1.1 Generic Letter 89-10 Program 4.1.2 Generic Letter 95-07 Program 6.5 Standby Liquid Control System
IV.1.B.i	Stresses	Attachment 5	3.2 Reactor Vessel 3.2.2 Reactor Vessel Structural Evaluation 3.4 Flow-Induced Vibration 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2 Balance-of-Plant Piping Evaluation
IV.1.B.ii	Cumulative usage factors	Attachment 5	3.2.2 Reactor Vessel Structural Evaluation, Table 3-5
IV.1.B.iii	Flow induced vibration	Attachment 1 Attachment 5 Attachment 10	3.5.2 Adverse Flow Effects 3.4 Flow-Induced Vibration Westinghouse Electric Company (WEC), Peach Bottom Units 2 and 3 Steam Dryer Structural Analysis Results at MUR Conditions, dated October 27, 2016

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
IV.1.B.iv	Changes in temperature (pre- and post-uprate)	Attachment 5	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.v	Changes in pressure (pre- and post- uprate)	Attachment 5	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vi	Changes in flow rate (pre- and post-uprate)	Attachment 5	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vii	High-energy line break locations	Attachment 5	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks
IV.1.B.viii	Jet impingement and thrust forces	Attachment 5	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks 10.1.2.7 Pipe Whip and Jet Impingement

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	Attachment 5	3.1 Nuclear System Pressure Relief/Overpressure Protection
IV.1.C.ii	Reactor vessel fluence evaluation	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure-temperature limit curves	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iv	Reactor vessel low temperature overpressure protection	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.v	Reactor vessel upper shelf energy	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	Attachment 5	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.D	Code of record	Attachment 5	3.2 Reactor Vessel 3.2.2 Reactor Vessel Structural Evaluation 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2 Balance-of-Plant Piping Evaluation
IV.1.E	Component inspection/testing programs and erosion/corrosion programs	Attachment 5	3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2 Balance-of- Plant Piping Evaluation 10.6 Plant Life

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	NA	NA
V. Electrical Equipment Design			
V.1.A	Emergency diesel generators	Attachment 5	6.1 AC Power 6.1.2 On-Site Power
V.1.B	Station blackout equipment	Attachment 5	9.3.2 Station Blackout
V.1.C	Environmental qualification of electrical equipment	Attachment 5	10.3 Environmental Qualification 10.3.1 Electrical Equipment
V.1.D	Grid stability	Attachment 5	6.1 AC Power 6.1.1 Off-Site Power
		Attachment 12	PJM Interconnection document, "Generator Transient Stability Study for Peach Bottom Atomic Power Station," and PECO document, Power Grid Voltage Analysis - Power Uprate Scenario for Peach Bottom Atomic Power Station."

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
VI. System Design			
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling, as applicable)	Attachment 5	3.4 Flow-Induced Vibration 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2 Balance-of-Plant Piping Evaluation 3.6 Reactor Recirculation System 3.7 Main Steam Line Flow Restrictors 3.8 Main Steam Isolation Valves 3.9 Reactor Core Isolation Cooling 3.10 Residual Heat Removal System 3.11 Reactor Water Cleanup System 4.2.5 ECCS Net Positive Suction Head
VI.1.B	Containment Systems	Attachment 5	4.1 Containment System Performance 4.1.1 Generic Letter 89-10 Program 4.1.2 Generic Letter 95-07 Program 4.1.3 Generic Letter 96-06
VI.1.C	Safety-related cooling water systems	Attachment 5	6.4 Water Systems 6.4.1 Service Water Systems 6.4.5 Reactor Building Closed Cooling Water System 6.4.6 Emergency Heat Sink

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
VI.1.D	Spent fuel pool storage and cooling systems	Attachment 5	6.3 Fuel Pool 6.3.1 Fuel Pool Cooling 6.3.2 Crud Activity and Corrosion Products 6.3.3 Radiation Levels 6.3.4 Fuel Racks
VI.1.E	Radioactive waste systems	Attachment 5	4.5 Standby Gas Treatment System 8.1 Liquid and Solid Waste Management 8.2 Gaseous Waste Management 8.3 Radiation Sources in the Reactor Core 8.4 Radiation Sources in Reactor Coolant 8.4.1 Coolant Activation Products 8.4.2 Activated Corrosion Products 8.4.3 Fission Products 8.5 Radiation Levels 8.6 Normal Operation Off-Site Doses
VI.1.F	Engineered safety features (ESF) heating, ventilation, and air conditioning systems	Attachment 5	4.4 Main Control Room Atmosphere Control System 4.7 Post-LOCA Combustible Gas Control System 6.6 Power-Dependent Heating, Ventilation and Air Conditioning

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
VII. Other			
VII.1	Operator actions and effects on time available	Attachment 5	4.1 Containment System Performance 6.7 Fire Protection 9.3 Special Events 9.3.1 Anticipated Transient Without Scram 9.3.2 Station Blackout 10.5 Operator Training and Human Factors
VII.2.A	Emergency and abnormal operating procedures	Attachment 5	10.9 Emergency Operating Procedures
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	Attachment 1	3.3.4 Disposition of NRC Criteria for Use of LEFM Topical Reports 3.3.6 Reactor Power Monitoring 3.5.6 Operator Training, Human Factors, and Procedures
		Attachment 5	10.5 Operator Training and Human Factors
VII.2.C	The control room plant reference simulator	Attachment 1	3.5.6 Operator Training, Human Factors, and Procedures
		Attachment 5	10.5 Operator Training and Human Factors
VII.2.D	The operator training program	Attachment 1	3.5.6 Operator Training, Human Factors, and Procedures
		Attachment 5	10.5 Operator Training and Human Factors
VII.3	Modifications completion	Attachment 1	3.5.3 Plant Modifications
VII.4	Procedure Revisions – Licensed Power Level	Attachment 1	3.3.6 Reactor Power Monitoring
		Attachment 1	3.5.6 Operator Training, Human Factors, and Procedures
		Attachment 5	10.5 Operator Training and Human Factors

NRC RIS 2002-03		LAR DOCUMENT	
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**
VII.5.A	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on types and amounts of effluents released offsite, and whether bounded by final environmental statement and previous Environmental Assessments for the plant	Attachment 1 Attachment 5	5.0 Environmental Consideration 8.0 Radwaste and Radiation Sources
VII.5.B	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on individual and cumulative occupational radiation exposure	Attachment 1 Attachment 5	5.0 Environmental Consideration 8.5 Radiation Levels
VIII.	**Changes to Technical Specifications, Protection System Settings, Emergency System Settings**		
VIII.1	A detailed discussion of each change to the plant's Technical Specifications, protection system settings, and/or emergency system settings needed to support the power uprate	Attachment 1 Attachment 2	2.0 Detailed Description All Markup of Proposed Operating License and Technical Specifications Pages
VIII.1.A	Description of the change	Attachment 1 Attachment 2	2.0 Detailed Description All Markup of Proposed Operating License and Technical Specifications Pages
VIII.1.B	Identification of analyses affected by and/or supporting the change	Attachment 1 Attachment 5	3.4 Evaluation of Changes to License and Technical Specifications All GE Hitachi Nuclear Energy Safety Analysis Report for Peach Bottom Atomic Power Station, Units 2 and 3 Thermal Power Optimization, NEDC-33873P

ATTACHMENT 4
NRC Regulatory Issue Summary 2002-03 Cross-Reference

NRC RIS 2002-03		LAR DOCUMENT		
Item No.	**DESCRIPTION**	**ATTACHMENT**	**SECTION and TITLE**	
VIII.1.C	Justification for the change, including the type of information discussed in Section III, above, for any analyses that support and/or are affected by change	Attachment 1	3.4	Evaluation of Changes to License and Technical Specifications
		Attachment 5	All	GEH Nuclear Energy Safety Analysis Report for Limerick Generating Station, Units 1 and 2 Thermal Power Optimization, NEDC-33484P

ATTACHMENT 6

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

GE Hitachi Nuclear Energy and EPRI Affidavits Supporting Withholding Attachments 5 and 14

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am the Vice President, Regulatory Affairs, Fuel 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 proprietary report NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization," Revision 0, dated February 2017. GEH proprietary information in NEDC-33873P is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]]. GEH proprietary information in figures and large objects is identified by double square brackets before and after the object. In each case, 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 U.S.C. §552(b)(4), and the Trade Secrets Act, 18 U.S.C. §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, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. 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 a license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its 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 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 for proprietary or confidentiality agreements or both 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.

(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) is classified as proprietary because it contains the detailed GEH methodology for thermal power optimization for GEH Boiling Water Reactors (BWRs). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute major GEH assets.

(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 is true and correct.

Executed on this 15th day of February 2017.

James F. Harrison

James F. Harrison
Vice President, Fuel Licensing
GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401
James.Harrison@ge.com

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am the Vice President, Regulatory Affairs, Fuel 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 the redline strikeout/blue additions version of GEH proprietary report NEDC-33873P, "Safety Analysis Report for Peach Bottom Atomic Power Station Thermal Power Optimization," Revision 0 - Redline Strikeout/Blue Additions Version, dated February 2017. GEH proprietary information in NEDC-33873P redline strikeout/blue additions version is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]]. GEH proprietary text may be in either red or blue font. GEH proprietary information in figures and large objects is identified by double square brackets before and after the object. In each case, 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 U.S.C. §552(b)(4), and the Trade Secrets Act, 18 U.S.C. §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, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. 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 a license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its 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 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 for proprietary or confidentiality agreements or both 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.

(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) is classified as proprietary because it contains the detailed GEH methodology for thermal power optimization for GEH Boiling Water Reactors (BWRs). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute major GEH assets.

(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 is true and correct.

Executed on this 15th day of February 2017.

James F. Harrison

James F. Harrison
Vice President, Fuel Licensing
GE-Hitachi Nuclear Energy Americas, LLC
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Wilmington, NC 28401
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EPRI | ELECTRIC POWER RESEARCH INSTITUTE

NEIL WILMSHURST
Vice President and
Chief Nuclear Officer

Ref. EPRI Project Number 669

January 10, 2017

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Information Included in:

Exelon, GE Hitachi Nuclear Energy Report titled:
"Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization" NEDC-33873P, Revision 0, dated February 2017

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Report and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Proprietary Information in confidence to assist the NRC review of the enclosed submittal to the NRC by Exelon. The Proprietary Information is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Proprietary Information provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 595-2732. Questions on the content of the Report should be directed to Andy McGehee of EPRI at (704) 502-6440.

Sincerely,

Attachment(s)
c: Sheldon Stuchell, NRC (sheldon.stuchell@nrc.gov)

ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

Exelon, GE Hitachi Nuclear Energy Report titled:
"Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization" NEDC-33873P, Revision 0, dated February 2017

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 1300 W WT Harris Blvd, Charlotte, NC. ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI Proprietary Information is identified in the above referenced report by double braces. Examples of such identification is as follows:

[[This sentence is an example.(E)]]

Tables containing EPRI Proprietary Information are identified with double brackets before and after the object. In each case the superscript notation (E) refers to this affidavit and all the bases included below, which provide the reasons for the proprietary determination.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information (see e.g., 10 C.F.R. § 2.390(a)(4):

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information.

c. The information sought to be withheld is considered to be proprietary for the following reasons. EPRI made a substantial economic investment to develop the Proprietary Information and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

d. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

> "'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

e. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

f. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of North Carolina.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 1-10-2017

[signature]

Neil M. Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 10th day of January, 2017 by Neil M. Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2021

DEBORAH H. ROUSE
NOTARY PUBLIC
CABARRUS
COUNTY
NORTH CAROLINA

ATTACHMENT 7

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

GE Hitachi Nuclear Energy Safety Analysis Report for Peach Bottom Atomic Power Station Units 2 and 3 Thermal Power Optimization, NEDC-33873 (Non-Proprietary Version)

GE Hitachi Nuclear Energy

NEDO-33873
Revision 0
February 2017

Non-Proprietary Information - Class I (Public)

SAFETY ANALYSIS REPORT
FOR
PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3
THERMAL POWER OPTIMIZATION

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33873P, Revision 0, 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 [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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ACRONYMS AND ABBREVIATIONS

Term	Definition
1RPT	One Recirculation Pump Trip
2RPT	Two Recirculation Pump Trip
A	Amperes (or Amps)
ABSP	Automated Backup Stability Protection
AC	Alternating Current
ADS	Automatic Depressurization System
AL	Analytical Limit
ALARA	As Low as Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOR	Analysis-of-Record
APRM	Average Power Range Monitor
ART	Adjusted Reference Temperature
ARTS	Average Power Range Monitor, Rod Block Monitor, Technical Specifications
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
ATWSI	ATWS with Core Instability
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOC	Beginning-of-Cycle
BOP	Balance-of-Plant
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAP	Containment Accident Pressure
CB&I	Chicago Bridge and Iron
CDA	Confirmation Density Algorithm

Term	Definition
CF	Core Flow
cfm	Cubic Feet per Minute
ChF	Chemistry Factor
CFD	Condensate Filter/Demineralizer
CFR	Code of Federal Regulations
CLTP	Current Licensed Thermal Power
CLTR	NEDC-33004P-A, Constant Pressure Power Uprate
COLR	Core Operating Limits Report
CRCWS	Control Room Chilled Water System
CRD	Control Rod Drive
CRGT	Control Rod Guide Tube
CS	Core Spray
CSC	Containment Spray Cooling
CSS	Core Support Structure
CUF	Cumulative Usage Factor
CW	Chilled Water
DBA	Design Basis Accident
DC	Direct Current
DCWS	Drywell Chilled Water System
DL	Design Limit
DSS-CD	Detect and Suppress Solution - Confirmation Density
ECCS	Emergency Core Cooling System
ECT	Emergency Cooling Tower
EDG	Emergency Diesel Generator
EHC	Electrohydraulic Control
EHS	Emergency Heat Sink
EFPY	Effective Full Power Years
ELTR1	NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate
ELTR2	NEDC-32523P-A, Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate
EMA	Equivalent Margin Analysis

Term	Definition
EOC	End-of-Cycle
EOP	Emergency Operating Procedure
EOR	End-of-Rated
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
ES	Extraction Steam
ESW	Emergency Service Water
FAC	Flow-Accelerated Corrosion
FB	Flow Biased
FFWTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FLEX	Diverse and Flexible Coping Strategies
FPC	Fuel Pool Cooling
FPCCS	Fuel Pool Cooling and Cleanup System
FSSD	Fire Safe Shutdown
FW	Feedwater
FWHOOS	Feedwater Heater(s) Out-of-Service
GDC	General Design Criterion
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy Americas LLC
GESTAR II	General Electric Standard Application for Reactor Fuel
GL	Generic Letter
GNF	Global Nuclear Fuel – Americas LLC
gpm	Gallons per Minute
HCTL	Heat Capacity Temperature Limit
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HgA	Mercury (pressure) - Absolute
HPCI	High Pressure Coolant Injection

Term	Definition
HPSW	High Pressure Service Water
HSBW	Hot Shutdown Boron Weight
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBOT	Instantaneous Break Opening Time
ICF	Increased Core Flow
IGSCC	Intergranular Stress Corrosion Cracking
IORV	Inadvertent Opening of a Relief Valve
IPE	Individual Plant Examination
ISP	Integrated Surveillance Program
JPSL	Jet Pump Sensing Line
K_{Ic}	Fracture Toughness Stress Intensity for Crack Initiation
ksi	Kips per Square Inch
kV	Kilovolt
LAR	License Amendment Request
LBPCT	Licensing Basis Peak Cladding Temperature
LCO	Limiting Condition for Operation
LEFM	Leading Edge Flow Meter
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOFW	Loss-of-Feedwater
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LTP	Licensed Thermal Power
LTR	Licensing Topical Report
LTS	Long-Term Solution
M+LTR	MELLLA+ Licensing Topical Report
M+SAR	MELLLA+ Safety Analysis Report

Term	Definition
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELB	Moderate Energy Line Break
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MeV	Million Electron Volts
MFLCPR	Maximum Fraction of Limiting Critical Power Ratio
MFLPD	Maximum Fraction of Limiting Power Density
Mlbm/Mlb	Millions of Pounds
MOC	Middle-of-Cycle
MOP	Mechanical Overpower
MOV	Motor-Operated Valve
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSLBA	Main Steam Line Break Accident
MUR	Measurement Uncertainty Recapture
MVA	Megavolt Amperes
MWe	Megawatt(s)-Electric
MWt	Megawatt(s)-Thermal
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NPSHA	Net Positive Suction Head Available
$NPSHR_{eff}$	Net Positive Suction Head Required Effective
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
NUMARC	Nuclear Management and Resources Council
OFS	Orificed Fuel Support

Term	Definition
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
P (F/E)	Conditional Failure Probability
P/F	Power/Flow
P-T	Pressure-Temperature
PBAPS	Peach Bottom Atomic Power Station Units 2 and 3
PCS	Pressure Control System
PCT	Peak Cladding Temperature
PCLRT	Primary Containment Leak Rate Test
PF	Power Factor
PLU	Power Load Unbalance
PR	Pressure Regulator
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Failure Open
psi	Pounds per Square Inch
psia	Pounds per Square Inch – Absolute
psid	Pounds per Square Inch – Differential
psig	Pounds per Square Inch – Gauge
PUSAR	Power Uprate Safety Analysis Report
RAI	Request for Additional Information
RBCCW	Reactor Building Closed Cooling Water
RBM	Rod Block Monitor
RCF	Rated Core Flow
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RFP	Reactor Feedwater Pump
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference

Term	Definition
RIS	Regulatory Issue Summary
RLB	Recirculation Line Break
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RRC	Reactor Recirculation
RRS	Reactor Recirculation System
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
S_A	Alternating Stress
S_{AD}	Amplitude Discriminator Setpoint
SAG	Severe Accident Guideline
SBO	Station Blackout
SC	Safety Communication
SDC	Shutdown Cooling
SE	Safety Evaluation
SECY	Office of the Secretary of the Commission (NRC)
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SJAE	Steam Jet Air Ejector
SL	Safety Limit
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SPC	Suppression Pool Cooling
SR	Surveillance Requirement
SRLR	Supplemental Reload Licensing Report

Term	Definition
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
SSV	Spring Safety Valve
STP	Simulated Thermal Power
SW	Service Water
T-M	Thermal-Mechanical
TAF	Top of Active Fuel
TBCCW	Turbine Building Closed Cooling Water
TBCS	Turbine Bypass Control System
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TFSP	Turbine First-Stage Pressure
T/G	Turbine-Generator
TIP	Traversing In-Core Probe
TLAA	Time Limiting Aging Analysis
TLO	Two Loop Operation
TLTR	NEDC-32938P-A, Revision 2, Thermal Power Optimization Licensing Topical Report
Tmin	Minimum Stable Film Boiling Temperature
TOP	Thermal Overpower
TPO	Thermal Power Optimization
TR	Topical Report
TS	Technical Specification(s)
TSAR	Thermal Power Optimization Safety Analysis Report
TSV	Turbine Stop Valve
TTWBP	Turbine Trip with Bypass
UBPCT	Upper Bound Peak Cladding Temperature
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
VSF	Vortex Shedding Frequency
VWO	Valves Wide Open
WRNM	Wide Range Neutron Monitor

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify increasing the licensed thermal power (LTP) at Peach Bottom Atomic Power Station Units 2 and 3 (PBAPS) to 4,016 megawatts-thermal (MWt). The requested license power level is approximately 1.66% above the current licensed thermal power (CLTP) level of 3,951 MWt.

This thermal power optimization (TPO) safety analysis report (i.e., TSAR) follows the Nuclear Regulatory Commission (NRC)-approved format and content for boiling water reactor (BWR) TSARs as documented in licensing topical report (LTR) NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," called "TLTR" (Reference 1). In accordance with the outline of the TSAR in TLTR Appendix A, every safety issue that should be addressed in a plant-specific TPO licensing report is addressed in this report.

Plant-specific evaluations and analyses were performed using the current licensing basis which includes approved amendments for extended power uprate (EPU) (Reference 2) and maximum extended load line limit analysis plus (MELLLA+) (Reference 3). In many cases, these evaluations were previously performed at 102% of CLTP, and thus, upon confirmation of continued applicability, bound the TPO uprated conditions. Some analyses are performed at TPO rated thermal power (RTP) because an uncertainty factor is accounted for in the methods, or the additional 2% margin is neither required nor appropriate (e.g., reactor heat balance). In addition, some analyses, (e.g., anticipated transient without scram (ATWS)), are conservatively performed at the TPO bounding thermal power of 101.7% of CLTP (i.e., 4,018 MWt).

Only previously NRC-approved or industry-accepted methods were used for the analysis of accidents, transients, and special events. Applicability of computer codes used for plant-specific analyses at TPO RTP for PBAPS is addressed in Section 1.2.2. Also, event and analysis descriptions that are provided in other licensing documents or the updated final safety analysis report (UFSAR) are not repeated. This report summarizes the results of the safety evaluations needed to justify a license amendment to allow for TPO operation.

The TLTR addresses power increases of up to 1.5% of CLTP, which will produce up to an approximately 2% increase in steam flow to the turbine-generator (T/G). The amount of power uprate (≤ 1.5%) contained in the TLTR was based on the expected reduction in power level uncertainty with the instrumentation technology available in 1999. The present instrumentation technology has evolved to where a power level uncertainty is reduced to as low as 0.3%, thereby supporting the evaluation of a power level increase of up to 1.7%. The safety evaluation for the TLTR, states, in Section 4.1, "However, plant-specific applications could request a higher TPO uprate (e.g., 1.7 percent), depending on the plant-specific feedwater flow measurement uncertainty."

A higher steam flow is achieved by increasing the reactor power along the current rod and core flow (CF) control lines. A limited number of operating parameters are changed, some setpoints are adjusted and instruments are recalibrated. Plant procedures are revised, and tests similar to some of the original startup tests are performed.

Evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents (DBAs), and previous licensing evaluations were performed. This report demonstrates that PBAPS can safely operate at a power level of 4,016 MWt.

The following evaluations were conducted in accordance with the criteria of TLTR Appendix B:

- All safety aspects of the plant that are affected by a 65 MWt increase in the thermal power level were evaluated, including the nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems.
- Evaluations and reviews were based on licensing criteria, codes, and standards applicable to the plant at the time of the TSAR submittal. There is no change in the previously established licensing basis for the plant, except for the increased power level.
- Evaluations and/or analyses were performed using NRC-approved or industry-accepted analysis methods for the UFSAR accidents, transients, and special events affected by TPO.
- Evaluations and reviews of the NSSS systems and components, containment structures, and BOP systems and components show continued compliance to the codes and standards applicable to the current plant licensing basis (i.e., no change to comply with more recent codes and standards is proposed due to TPO).
- NSSS components and systems were reviewed to confirm that they continue to comply with the functional and regulatory requirements specified in the UFSAR and/or applicable reload license.
- PBAPS has previously installed the Caldon® Leading Edge Flow Meter (LEFM) Check Plus™ system. No modifications to the LEFM are needed for the TPO implementation.
- All plant systems and components potentially affected by an increased thermal power level were reviewed to ensure that there is no significant increase in challenges to the safety systems.
- A review was performed to ensure that the increased thermal power level continues to comply with the existing plant environmental regulations.
- Evaluations were performed to assess the operational conditions of PBAPS in the TPO expanded MELLLA+ operating domain to ensure key performance parameters are within the PBAPS operating experience base.
- An assessment, as defined in 10 Code of Federal Regulations (CFR) 50.92(c), was performed to establish that no significant hazards consideration exists as a result of operation at the increased power level.
- A review of the UFSAR and approved design changes ensures adequate evaluation of the licensing basis for the effect of TPO through the date of submittal.

The plant licensing requirements have been reviewed, and it is concluded that this TPO can be accommodated:

- Without a significant increase in the probability or consequences of an accident previously evaluated,
- Without creating the possibility of a new or different kind of accident from any accident previously evaluated, and
- Without exceeding any existing regulatory limits applicable to the plant, which might cause a significant reduction in a margin of safety.

Therefore, the requested TPO uprate does not involve a significant hazards consideration.

1.0 INTRODUCTION

1.1 OVERVIEW

This report addresses a TPO power uprate of 65 MWt, or approximately 1.66% of the CLTP, consistent with the magnitude of the thermal power uncertainty reduction for PBAPS. The reduced thermal power uncertainty will result in an increase in LTP from 3,951 MWt to 4,016 MWt.

This report follows the NRC-approved format and content for a BWR TSAR documented in NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization" (TLTR) (Reference 1). Power uprates in GE BWRs of up to 120% of original licensed thermal power (OLTP) are based on the generic guidelines and approach defined in the safety evaluation reports (SERs) provided in NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) (Reference 4) and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2) (Reference 5). Note that the PBAPS EPU was based on NEDC-33004P-A, "Constant Pressure Power Uprate," (CLTR) (Reference 6), and involved no change in reactor operating pressure.

Since their NRC approval, numerous EPU submittals have been based on these reports. The outline for the TSAR in TLTR Appendix A follows the same pattern as that used for the EPUs. All of the issues that should be addressed in a plant-specific TPO licensing report for PBAPS are included in this TSAR. Plant-specific evaluations and analyses are performed using the current licensing basis which includes approved amendments for EPU (Reference 2) and MELLLA+ (Reference 3).

The amount of power uprate (≤ 1.5%) discussed in the TLTR was based on the expected reduction in power level uncertainty consistent with the instrumentation technology available in 1999. The present instrumentation technology has evolved to where a power level uncertainty is reduced to as low as 0.3%, thereby supporting the evaluation of a power level increase of up to 1.7%. Section 4.1 of the safety evaluation for the TLTR, states, "However, plant-specific applications could request a higher TPO uprate (e.g., 1.7 percent), depending on the plant-specific feedwater flow measurement uncertainty."

BWR plants have already been authorized, in accordance with the TLTR, to increase their thermal power above the OLTP based on a reduction in the uncertainty in the determination of the power through improved feedwater (FW) flow rate measurements. When a previous uprate (other than a TPO) has been accomplished, such as the EPU in the case of PBAPS, the ≥ 102% safety analysis basis is reestablished above the uprated power level. Therefore, the uprated GEH BWR plants, which have the ≥ 102% safety analysis basis, have the capability to implement a subsequent TPO uprate.

For a plant that has already implemented an EPU, a TPO uprate application will rely on the TLTR approach in terms of topics identified as in-scope and the disposition of those topics. Consequently, for plants that have implemented previous power uprates, the generic dispositions in the TLTR require a plant-specific evaluation and justification in order to assess the effect of a TPO uprate as stated in TLTR Section 4. Thus, plants seeking to apply a TPO uprate to a previous uprate that would result in LTP in excess of 120% of OLTP, must provide plant-specific evaluations for those evaluations not performed at 102% of CLTP.

1.2 PURPOSE AND APPROACH

1.2.1 TPO Analysis Basis

PBAPS was originally licensed at 3,293 MWt and has received power uprates via amendments to the facility operating license. PBAPS was uprated to the CLTP level of 3,951 MWt through the issuance of an amendment for EPU (Reference 2). Where required, the current safety analysis basis assumes that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level. The analyses performed at 102% of CLTP (i.e., 4,030 MWt) remain applicable at the TPO RTP, because the 2% factor from Regulatory Guide (RG) 1.49, "Power Levels of Nuclear Power Plants," (Reference 7) bounds the improvement in the FW flow measurements. Some analyses are performed at TPO RTP because the uncertainty factor is accounted for in the methods, or the additional 2% margin is neither required nor appropriate (e.g., reactor heat balance). In addition, some analyses (e.g., ATWS) are conservatively performed at the TPO bounding thermal power of 101.7% of CLTP (i.e., 4,018 MWt). Detailed descriptions of the basis for the TPO analyses are provided in the subsequent sections of this report.

Figure 1-1a and Figure 1-1b illustrate the TPO power/flow (P/F) operating map for the analysis at 4,016 MWt, or approximately 101.66% of CLTP, for PBAPS. The approach to achieve a higher thermal power level is to increase CF along the established MELLLA+ boundary. This strategy allows PBAPS to maintain most of the existing available CF operational flexibility while assuring that low power-related issues (e.g., stability and ATWS instability) do not change because of the TPO uprate.

The TPO uprated power domain is established by extending the current MELLLA+ upper boundary with no increase in the maximum CF. The MELLLA+ domain for TPO includes the region between the CLTP MELLLA+ and maximum extended load line limit analysis (MELLLA) boundaries and the new additional region between the MELLLA+ and MELLLA boundaries extended to TPO power. Part of the MELLLA+ domain extends into the increased core flow (ICF) region. When end of full power reactivity condition (all-rods-out) is reached, end-of-cycle (EOC) coast down may be used to extend the power generation period. Previously licensed performance improvement features are presented in Section 1.3.2.

With respect to absolute thermal power and flow, there is no change in the extent of the single loop operating (SLO) domain as a result of the TPO uprate. PBAPS is not allowed to operate in SLO in the MELLLA+ operating domain. For PBAPS, the maximum analyzed reactor core thermal power for SLO in the MELLLA domain remains at the licensed limit. Therefore, the SLO analyses are not provided.

PBAPS has implemented the detect and suppress solution – confirmation density (DSS-CD) long-term solution (LTS) with implementation of the MELLLA+ license amendment

(Reference 3) and consistent with Reference 8. [[

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The TPO uprate is accomplished with no increase in the nominal vessel dome pressure. This minimizes the effect of uprating on reactor thermal duty, evaluations of environmental conditions, and minimizes changes to instrument setpoints related to system pressure. Satisfactory reactor pressure control capability is maintained by evaluating the steam flow margin available at the turbine inlet. This operational aspect of the TPO uprate will be demonstrated by performing controller testing as described in Section 10.4.

The general operational conditions for PBAPS in the TPO expanded MELLLA+ operating domain (122% of OLTP) are within expected parameters of the PBAPS operating experience base and therefore acceptable in addressing continued applicability of GEH methods to the PBAPS TSAR analyses.

The fuel properties for all analyses included in this TSAR are based on the PRIME methodology (Reference 9), except for the loss-of-feedwater (LOFW) event, which is based on an existing analysis using GSTRM (Reference 10). An evaluation of the PB TPO LOFW event is discussed in Section 3.9.

This report also addresses continued applicability at TPO RTP conditions, which includes EPU and MELLLA+, of the limitations and conditions described in the following NRC SERs:

- The NRC SER for GEH LTR NEDC-33006P-A, "Maximum Extended Load Line Limit Analysis Plus," referred to as the M+LTR (Reference 11);
- The NRC SER for GEH LTR NEDC-33173P-A, "Applicability of GE Methods to Expanded Operating Domains," referred to as the Methods LTR (Reference 12);
- The NRC SER for GEH LTR NEDC-33075P, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," referred to as the DSS-CD LTR (Reference 8); and
- The NRC SER for GEH LTR NEDE 32906P, Supplement 3-A, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients" (Reference 13).

A complete listing of the limitations and conditions required in the M+LTR SER, Methods LTR SER, DSS-CD LTR SER, and Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients LTR SER is presented in Appendices A, B, C, and D, respectively. Consistent with M+LTR SER Limitation and Condition 12.2, the disposition of each applicable limitation and condition is addressed in these appendices. In many cases, information showing compliance to a limitation and condition from the PBAPS power uprate safety analysis report (PUSAR) (Reference 14) or MELLLA+ safety analysis report (M+SAR) (Reference 15) remains applicable at TPO RTP conditions. In such cases, references to the relevant sections of Reference 14 or Reference 15 are provided.

Additionally, as required by M+LTR SER Limitation and Condition 12.4, because PBAPS is already operating with an EPU, the supplemental reload licensing report (SRLR) for the initial TPO implementation cycle will be submitted for NRC staff confirmation.

1.2.2 Margins

Factors and margins specified by the application of design code rules are maintained, as are other margin-assuring acceptance criteria used to judge the acceptability of the plant. NRC-approved or industry-accepted computer codes and calculation techniques are used in the safety analyses for the TPO uprate. A list of the NSSS computer codes used in the evaluations is provided in Table 1-1a. Computer codes used in previous analyses (i.e., analyses at 102% of CLTP, which includes EPU and MELLLA+) are not listed. Applicability of computer codes listed in Table 1-1a used for plant-specific analyses at TPO RTP for PBAPS is addressed in Table 1-1b.

1.2.3 Scope of Evaluations

Plant-specific evaluations, using the methodology discussed in TLTR Appendix B, are performed using the current licensing basis which includes approved amendments for EPU (Reference 2) and MELLLA+ (Reference 3). As required by M+LTR SER Limitation and Condition 12.3.a, the plant-specific evaluations are reported consistent with the content, structure, and level of detail indicated in the M+LTR, with adjustments for TPO conditions and TSAR reporting structure as defined in the TLTR.

The scope of the evaluations is summarized in the following sections:

2.0 Reactor Core and Fuel Performance

Overall heat balance and power-flow operating map information are provided. Key core performance parameters are evaluated on an equilibrium cycle for PBAPS in the TPO expanded MELLLA+ operating domain and demonstrated to be within expected parameters of the PBAPS operating experience base. As required by M+LTR SER Limitation and Condition 12.4, the reload licensing evaluation will continue to be evaluated at TPO conditions and confirmed acceptable for each cycle, in accordance with the General Electric Standard Application for Reactor Fuel (GESTAR-II, Reference 16) requirements, and reported in the SRLR.

3.0 Reactor Coolant and Connected Systems

Evaluations of the NSSS components and systems are performed at the TPO conditions. These evaluations confirm the acceptability of the TPO changes in process variables in the NSSS.

4.0 Engineered Safety Features

The effects of TPO changes on the containment, emergency core cooling system (ECCS), standby gas treatment system (SGTS), and other engineered safety features are evaluated for key events. The evaluations include the containment responses during limiting abnormal events, loss-of-coolant accidents (LOCAs), and safety relief valve (SRV) containment dynamic loads.

5.0 Instrumentation and Control

The instrumentation and control signal ranges and analytical limits (ALs) for setpoints are evaluated to establish the effects of TPO changes in process parameters. If required, analyses are performed to determine the need for setpoint changes for various functions. In general, setpoints are changed only to maintain adequate operating margins between plant operating parameters and trip values.

6.0 Electrical Power and Auxiliary Systems

Evaluations are performed to establish the operational capability of the plant electrical power and distribution systems and auxiliary systems to ensure that they are capable of supporting safe plant operation at the TPO RTP level.

7.0 Power Conversion Systems

Evaluations are performed to establish the operational capability of various (non-safety-related) BOP systems and components to ensure that they are capable of delivering the increased TPO power output.

8.0 Radwaste and Radiation Sources

The liquid and gaseous waste management systems are evaluated at TPO conditions to show that applicable release limits continue to be met during operation at the TPO RTP level. The radiological consequences are evaluated to show that applicable regulations are met for TPO including the effect on source terms, on-site doses, and off-site doses during normal operation.

9.0 Reactor Safety Performance Evaluations

[[

]] The standard reload analyses consider the plant conditions for each fuel cycle.

10.0 Other Evaluations

High energy line break (HELB) and environmental qualification (EQ) evaluations are performed at bounding conditions for the TPO range to show the continued operability of plant equipment under TPO conditions. The individual plant examination (IPE) probabilistic risk assessment (PRA) will not be updated, because the change in plant risk from the subject power uprate is insignificant. This conclusion is supported by NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 17).

1.2.4 Exceptions to the TLTR

No exceptions are requested to the TLTR because this evaluation follows the protocol as approved by the NRC. Because PBAPS has implemented a previous power uprate to 120% of OLTP, no generic evaluations in the TLTR are utilized. Plant-specific evaluations and analyses, consistent with approaches used in the TLTR and guidance in Section 4.2 of the TLTR safety evaluation, have been performed.

1.2.5 Concurrent Changes Unrelated to TPO

No concurrent changes unrelated to TPO are included in this evaluation.

1.3 TPO PLANT OPERATING CONDITIONS

1.3.1 Reactor Heat Balance

The nominal heat balance diagram at TPO conditions is presented in Figure 1-2.

The small changes in thermal-hydraulic parameters for TPO are identified in Table 1-2. These parameters are generated for TPO by performing reactor heat balances that relate the reactor thermal-hydraulic parameters to the increased plant FW and steam flow conditions. Input from PBAPS operation (e.g., steam line pressure drop) is considered to match expected TPO uprate conditions.

In accordance with M+LTR SER Limitation and Condition 12.5.c, PBAPS will include the P/F map in the Core Operating Limits Report (COLR) once the MELLLA+ operating domain with TPO expanded region is approved.

PBAPS continues to exceed the power-to-flow ratio of 50 MWt/Mlbm/hr at 55% CF in the lower portion of the MELLLA+ operating domain. This region of the MELLLA+ operating domain is not affected by the implementation of TPO. The previous assessment of the limitation with respect to the conservatism of the power distribution uncertainties as performed in Section 2.2.5 of the M+SAR (Reference 15) continues to apply for TPO. The results of this assessment are provided in TSAR Section 2.2.5.

1.3.2 Reactor Performance Improvement Features

The following performance improvement and equipment out-of-service (OOS) features currently licensed at PBAPS are acceptable at the TPO RTP level. As required by M+LTR SER Limitation and Condition 12.5.a, those features prohibited in the MELLLA+ domain, including as expanded by TPO conditions, are so indicated:

Performance Improvement Feature
MELLLA+ (85.2% of Rated Core Flow at TPO RTP)
ICF (110.0% of rated)
Feedwater Heater(s) OOS (FWHOOS), 55ºF Reduction (not allowed in MELLLA+ domain)
FWHOOS, 10ºF Reduction (allowed in MELLLA+ domain per Operating License Condition 2.C(16))
Final Feedwater Temperature Reduction (FFWTR), 90ºF Reduction (not allowed in MELLLA+ domain)

Performance Improvement Feature
SLO (not allowed in MELLLA+ domain)
Turbine Bypass Valve (TBV) OOS
One SRV OOS
Recirculation Pump Trip (RPT) OOS
Turbine Stop Valve (TSV) / Turbine Control Valve (TCV) OOS
Main Steam Isolation Valve (MSIV) OOS (≤ 75% of 3,514 MWt)
Pressure Regulator (PR) OOS
Power Load Unbalance (PLU) OOS
Average Power Range Monitor, Rod Block Monitor, Technical Specifications (ARTS) Program
24 Month Cycle

1.4 BASIS FOR TPO UPRATE

The safety analyses in this report are based on a total thermal power measurement uncertainty of 0.34%. The detailed basis for this uncertainty value is provided in PBAPS calculations ER-463 (Reference 18) and ER-464 (Reference 19), which addresses the improved FW flow measurement accuracy using the Caldon® Leading Edge Flow Meter Check Plus™ system.

1.5 SUMMARY AND CONCLUSIONS

This report has evaluated a TPO power uprate of up to 65 MWt, or approximately 1.66% of CLTP. Plant licensing challenges have been reviewed. Table 1-3 demonstrates the TPO uprate can be accommodated without:

- A significant increase in the probability or consequences of an accident previously evaluated;
- Creating the possibility of a new or different kind of accident from any accident previously evaluated; and,
- Exceeding any existing regulatory limits or design allowable limits applicable to the plant which might cause a reduction in a margin of safety.

The TPO uprate described herein thus involves no significant hazards consideration.

Table 1-1a Computer Codes for TPO Analyses

Task	Computer Code*	Version or Revision	NRC Approved	Comments
Reactor Recirculation System	BILBO	04V	(1)	NEDE-23504, February 1977
Nominal Reactor Heat Balance	ISCOR	09	Y (2)	NEDE-24011P Rev. 0 SER
Reactor Internal Pressure Differences	ISCOR	09	Y (2)	NEDE-24011P Rev. 0 SER
Station Blackout	SHEX	06	Y (3)	
Reactor Core and Fuel Performance	TGBLA PANAC ISCOR	06 11 09	Y (4) Y (4) Y (2)	NEDE-30130P-A NEDE-30130P-A NEDE-24011P Rev. 0 SER
Thermal-Hydraulic Stability	ODYSY ISCOR PANAC TRACG	05 09 11 04	Y Y (2) Y (4) Y	NEDE-33213P-A NEDE-24011P Rev. 0 SER NEDE-30130-A NEDE-33147P-A Rev. 4
Piping Components Flow Induced Vibration (FIV)	SAP4G07P	07	(1)	
Anticipated Transient Without Scram	ODYN STEMP PANACEA ISCOR TRACG TASC	10 04 11 09 04 03	Y (1) Y (4) Y (2) Y Y	NEDE-24154P-A Supplement 1, Vol. 4 NEDE-30130-P-A NEDE-24011-P Rev. 0 SER NEDE-32906P Supplement 3-A, Rev. 1 NEDC-32084P-A Rev. 2
Anticipated Transient Without Scram with Instability	TRACG	04	Y (5)	
Appendix R Fire Protection	SHEX SAFER PRIME	06 04 03	Y (3) Y (6,7) Y (8)	

* The application of these codes to the PBAPS MELLLA+ and TPO analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the MELLLA+ and TPO programs.

Notes are on the next page.

Notes for Table 1-1a:

(1) Not a safety analysis code that requires NRC approval. The code application is reviewed and approved by GEH for "Level 2" application and is part of GEH's standard design process. The application of this code has been used in previous power uprate submittals.

(2) The ISCOR code is not approved by name. However, in the SER supporting approval of NEDE-24011P Revision 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE), the NRC finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, transient, ATWS, stability, reactor core and fuel performance, and LOCA applications is consistent with the approved models and methods.

(3) The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to Gary L. Sozzi (GE) from Ashok Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ American Nuclear Society (ANS) 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 20).

(4) The use of TGBLA Version 06 and PANAC Version 11 was initiated following approval of Amendment 26 of GESTAR II by letter from S. A. Richards (NRC) to G. A. Watford (GE) Subject: "Amendment 26 to GE LTR NEDE-24011P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(5) The TRACG04 code is not approved by the NRC for long-term ATWS calculations including ATWS with depressurization and ATWS with core instability. However, the use of TRACG04 for the best-estimate TRACG ATWS analysis is consistent with the NRC safety evaluation (SE) for NEDC-33006P.

(6) General Electric Company, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, October 1987.

(7) Letter, Richard E. Kingston (GEH) to NRC, "Transmittal of Revision 1 of NEDC-32950, Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," MFN 07-406, July 31, 2007.

(8) Application of PRIME models and data to downstream methods is approved by NEDO-33173 Supplement 4-A, "Implementation of PRIME Models and Data in Downstream Methods," Revision 1, November 2012.

Table 1-1b Applicability of Computer Codes at TPO Conditions

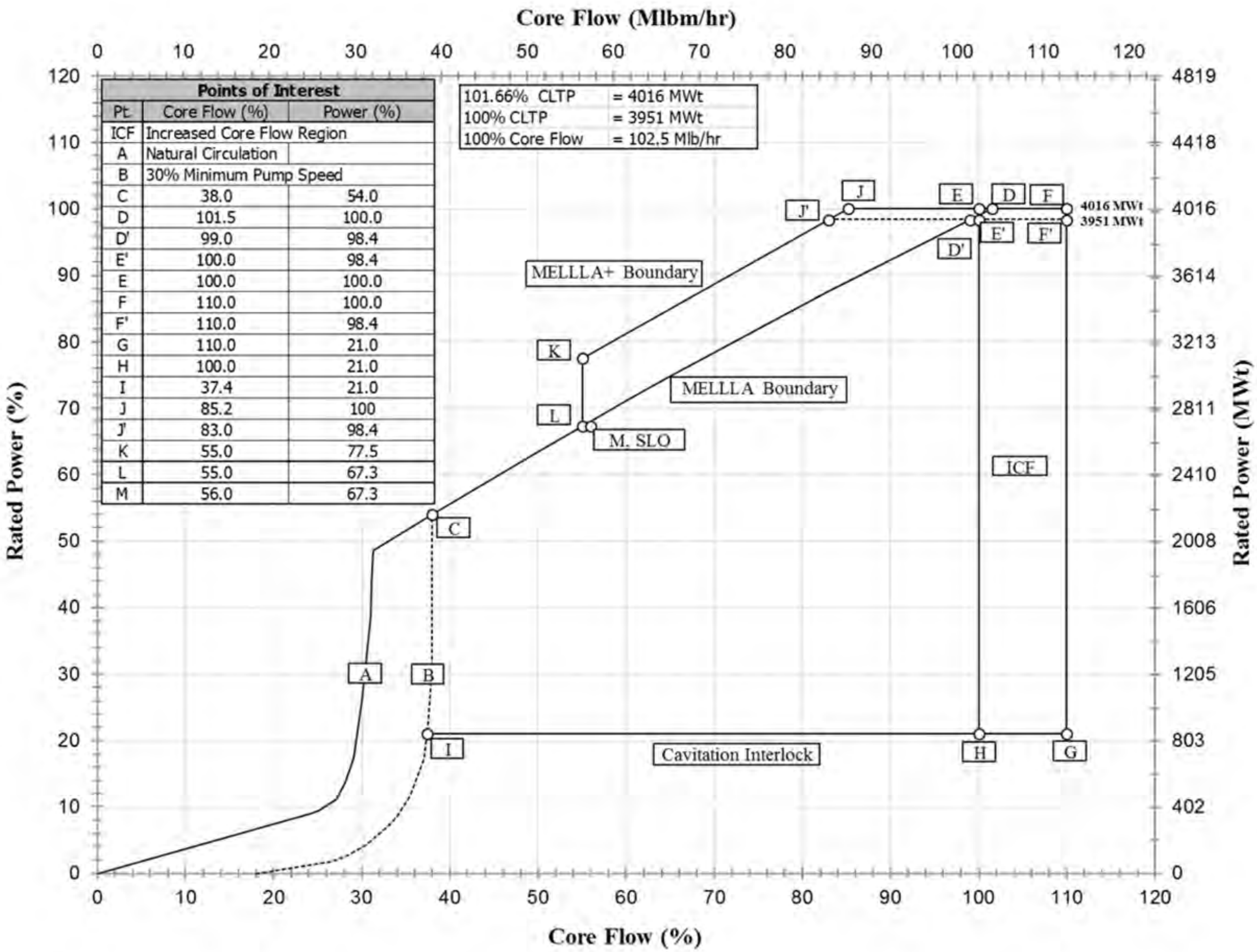
Computer Code	Version or Revision	Applicability Statement
BILBO	04V	1
ISCOR	09	1, 2
ODYN	10	2
ODYSY	05	1
PANAC	11	1, 2
PRIME	03	1
SAP4G07P	07	1
SAFER	04	1
SHEX	06	1
STEMP	04	2
TASC	03	2
TGBLA	06	1, 2
TRACG	04	1, 2, 3
Applicability Statements		
1. These codes have no inherent limitations related to power level. Any BWR operating condition can be simulated with the proper choice of convergence criteria coupled with appropriate hydraulic correlations for the given conditions. 2. These codes are applicable to PBAPS TPO RTP conditions, which include EPU and MELLLA+, because they remain within the qualification basis for each of the codes. The small change in power does not disqualify the codes to calculate the parameters they were designed to calculate, and results show that the conclusions made from the EPU and MELLLA+ projects (References 14 and 15) are not affected. The slightly higher power and steam flow results in slightly higher vessel pressure and containment temperature/pressure, but there is still margin to the limits. 3. For TRACG ATWS with core instability (ATWSI) calculations, the conditions following a recirculation pump trip are the same as compared to the MELLLA+ analysis (Reference 15) because both initiate from the same MELLLA+ rod line.		

Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions

Parameter	CLTP	TPO RTP
Thermal Power (MWt) (Percent of Current Licensed Power)	3,951 100.0	4,016 101.66
Steam Flow (Mlbm/hr) (Percent of Current Rated)	16.171 100.0	16.476 101.9
FW Flow (Mlbm/hr) (Percent of Current Rated)	16.139 100.0	16.444 101.9
Dome Pressure (psia)	1,050	1,050
Dome Temperature (°F)	550.6	550.6
FW Temperature (°F)	381.5	383.4
Full Power Core Flow Range (Mlbm/hr) (Percent of Current Rated)	85.1 to 112.8 (83 to 110)	87.3 to 112.8 (85.2 to 110)

Table 1-3 Summary of Effect of TPO Uprate on Licensing Criteria

Key Licensing Criteria	Effect of TPO Thermal Power Increase	Explanation of Effect
LOCA challenges to fuel (10 CFR 50, Appendix K)	No increase in peak cladding temperature (PCT), no change of maximum linear heat generation rate (LHGR) required.	Previous analysis accounted for ≥ 102% of licensed power, bounding TPO operation. No vessel pressure increase.
Change of operating limit Minimum Critical Power Ratio (OLMCPR)	< 0.01 increase.	Minor increase (< 0.01) due to slightly higher power density and increased minimum critical power ratio (MCPR) safety limit (SL) (slightly flatter radial power distribution).
Challenges to reactor pressure vessel (RPV) overpressure	No increase in peak pressure.	No increase because previous analysis accounted for ≥ 102% overpower, bounding TPO operation.
Primary containment pressure during a LOCA	No increase in peak containment pressure.	Previous analysis accounted for 102% overpower, bounding TPO operation. No vessel pressure increase. No increase in energy to the suppression pool.
Suppression pool temperature during a LOCA	No increase in peak suppression pool temperature.	Previous analysis accounted for 102% overpower, bounding TPO operation. No vessel pressure increase. No increase in energy to the suppression pool.
Offsite radiation release, DBAs	No increase (remains within 10 CFR 50.67).	Previous analysis bounds TPO operation. No RPV pressure increase.
Onsite radiation dose, normal operation	PBAPS as low as reasonably achievable (ALARA) program controls compensate for any minor increase in radiation levels.	Slightly higher inventory of radionuclides in steam/FW flow paths.
Heat discharge to environment	Less than 1°F temperature increase.	Small (1.66%) power increase.
Equipment qualification	Remains within current pressure, radiation, and temperature envelopes.	The resulting environmental conditions are bounded by the existing environmental parameters specified for use in the EQ program.
Fracture toughness, 10 CFR 50, Appendix G	Less than 0.5°F increase in reference temperature of the nil-ductility transition (RT_{NDT}).	Small increase in neutron fluence.
Stability	No direct effect of TPO uprate because applicable stability regions and lines are extended beyond the absolute values associated with the current boundaries to preserve MWt-CF boundaries as applicable for each stability option.	No increase in maximum rod line boundary. Characteristics of each reload core continue to be evaluated as required for each stability option.
ATWS peak vessel pressure	Slight increase (11 psi) is within existing American Society of Mechanical Engineers (ASME) code "Emergency" category stress limit.	The increased pressure is due to a slightly increased power relative to SRV capacity.
Vessel and NSSS equipment design pressure	No change.	Comply with existing ASME code stress limits for all categories.



Points of Interest		
Pt.	Core Flow (%)	Power (%)
ICF	Increased Core Flow Region	
A	Natural Circulation	
B	30% Minimum Pump Speed	
C	38.0	54.0
D	101.5	100.0
D'	99.0	98.4
E'	100.0	98.4
E	100.0	100.0
F	110.0	100.0
F'	110.0	98.4
G	110.0	21.0
H	100.0	21.0
I	37.4	21.0
J	85.2	100
J'	83.0	98.4
K	55.0	77.5
L	55.0	67.3
M	56.0	67.3

Figure 1-1a Power/Flow Map for TPO

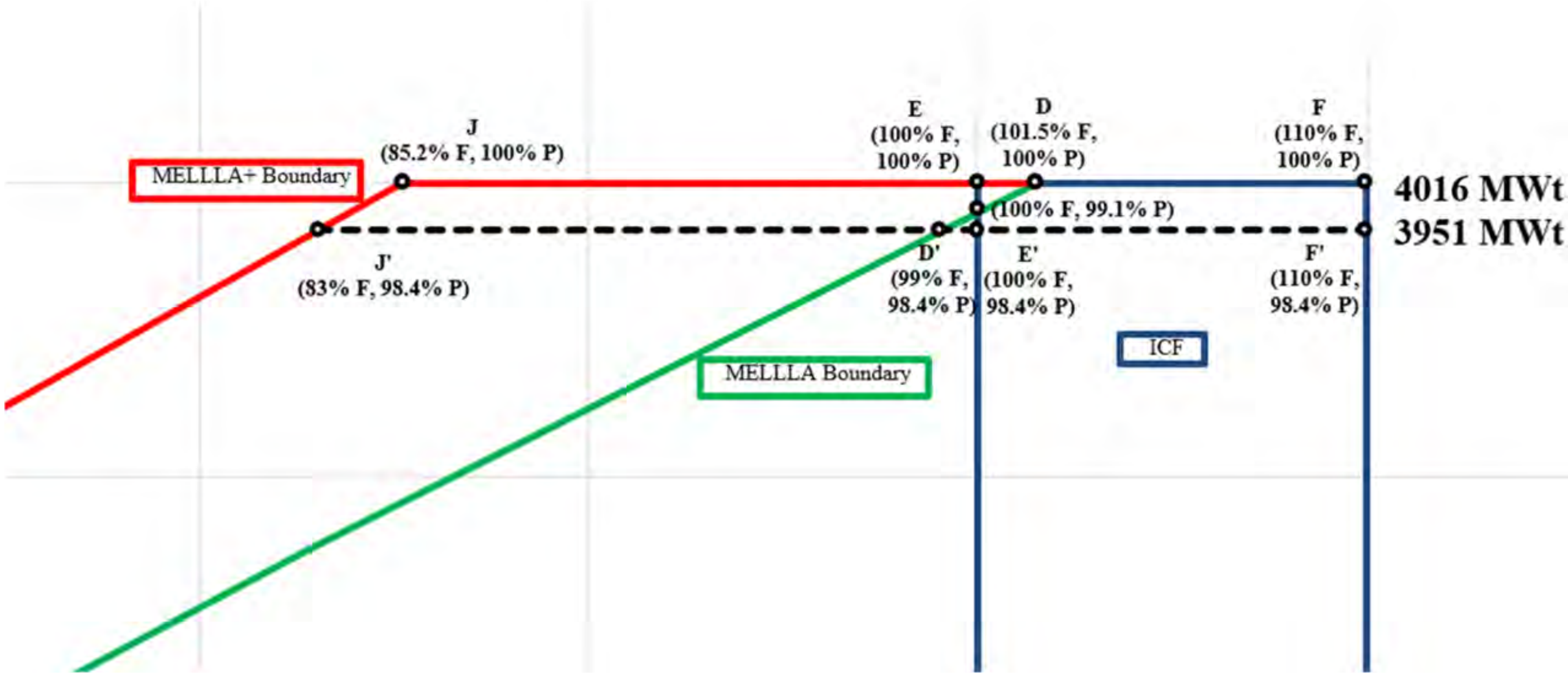
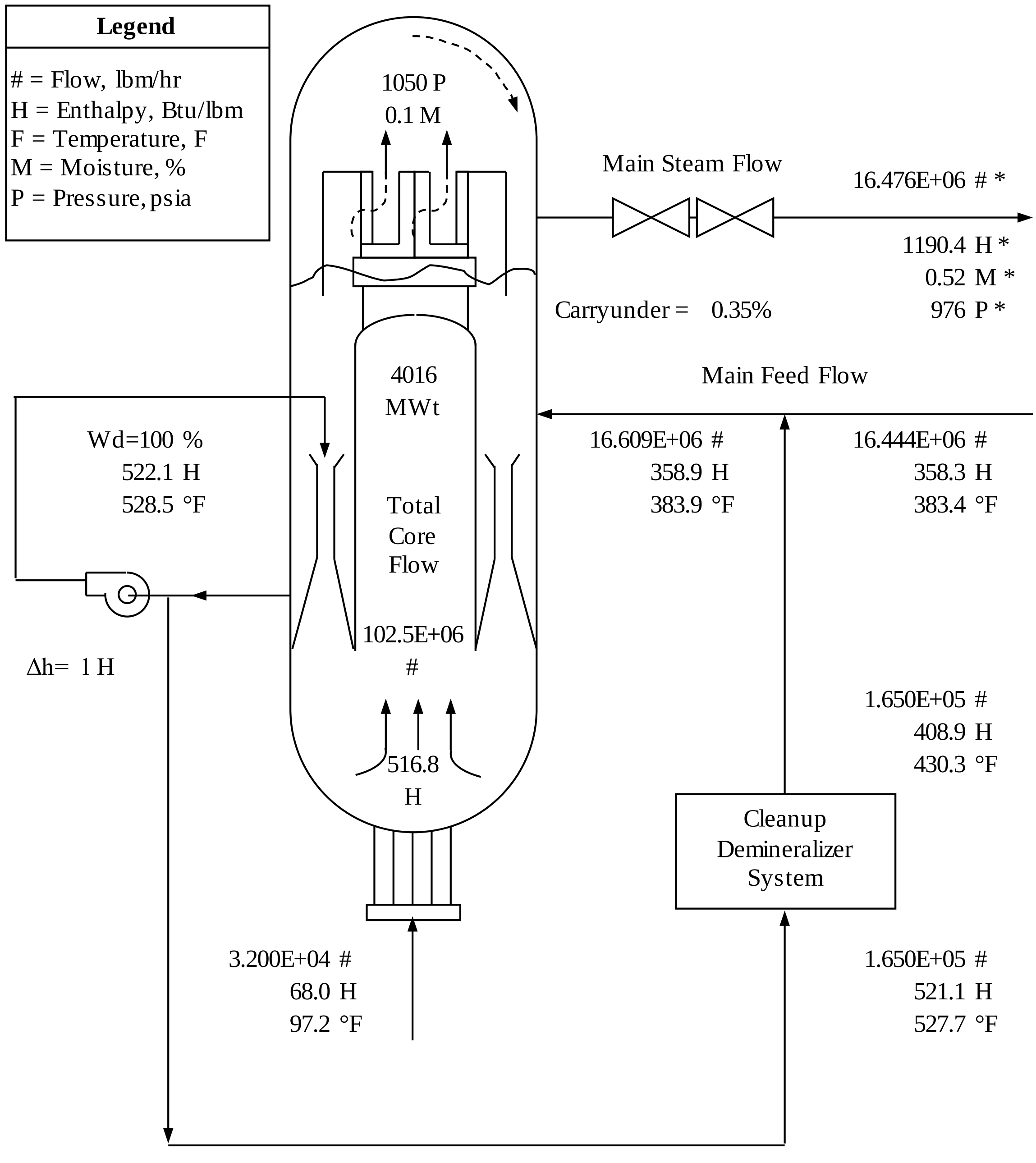


Figure 1-1b Power/Flow Map for TPO (Top Right Corner)



*Conditions at upstream side of TSV

Core Thermal Power	4016.0	
Pump Heating	10.5	
Cleanup Losses	-5.4	
Other System Losses	-0.6	
Turbine Cycle Use	4020.5	MWt

Figure 1-2 Reactor Heat Balance – TPO Power, 100% Core Flow

2.0 REACTOR CORE AND FUEL PERFORMANCE

This section addresses the evaluations that are applicable to MELLLA+, including the TPO domain expansion from 120% of OLTP (= CLTP) to 122% of OLTP (= 102% CLTP).

Because PBAPS currently uses Global Nuclear Fuel – Americas, LLC (GNF) fuel, type GNF2, the following limitations and conditions from the Methods LTR SER (Reference 12) are not applicable to the PBAPS TSAR (see Appendix A):

- Limitation and Condition 9.13: Application of 10 weight percent GD,
- Limitation and Condition 9.21: Mixed Core Method 1, and
- Limitation and Condition 9.22: Mixed Core Method 2.

2.1 FUEL DESIGN AND OPERATION

At the TPO RTP conditions, all fuel and core design limits (DLs) are met by the deployment of fuel enrichment and burnable poison, control rod pattern management, and CF adjustments. Revised loading patterns, slightly larger batch sizes, and potentially new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length. NRC-approved limits for burnup on the fuel are not exceeded. Therefore, the reactor core and current fuel design is adequate for TPO operation.

Figures 2-1 through 2-17 demonstrate that the general operational conditions for PBAPS in the TPO expanded MELLLA+ operating domain are within expected parameters of the PBAPS operating experience base and therefore acceptable in addressing continued applicability of GEH methods to the PBAPS TSAR analyses.

2.1.1 Fuel Product Line

The fuel design limits are established for all new fuel product line designs as a part of the fuel introduction and reload analyses. No changes in fuel product line design as a consequence of MELLLA+ or TPO are required. Because implementation of the extended MELLLA+ operating domain with TPO does not necessitate a new fuel design, no additional fuel and core design evaluations are required. However, the same fuel and core design evaluations performed in the PBAPS M+SAR are performed in the PBAPS TSAR to demonstrate that the change from 120% of OLTP to 122% of OLTP is inconsequential. The magnitude of changes to thermal margins and other core characteristics are within normal cycle-to-cycle variation due to changes in plant energy utilization plans as described in Section 5.7 of the TLTR SER (Reference 1).

PBAPS currently operates with GNF fuel. The PBAPS Unit 2 and Unit 3 cores at the time of TPO implementation are expected to consist only of GNF2 fuel. For PBAPS, no new fuel product line design is introduced and there is no change to fuel DLs required by the TPO introduction at PBAPS. Consistent with M+LTR SER Limitation and Condition 12.3.e, the use of GNF2 is specifically addressed in the M+SAR and TSAR.

The fuel product line design for PBAPS will be evaluated for the reload core prior to TPO implementation consistent with the GNF2 requirements (Reference 21).

2.1.2 Core Design

There is a small change to the average power density and average bundle power as a result of the TPO operating domain expansion, as allowed under TLTR SER Section 4.1. Thus, the fuel thermal monitoring threshold is adjusted as noted in Section 2.1.3. There are no changes to the PBAPS fuel or fuel DLs as a result of TPO. PBAPS continues to use the GNF2 fuel product line.

The TPO expanded MELLLA+ operating domain allows for higher bundle powers, but not lower bundle flows due to the extension of the existing MELLLA+ boundary. The bundle power to flow ratios at TPO 122% of OLTP core power and 85.2% CF conditions are less than the bundle power to flow ratios at the previous MELLLA+ 120% of OLTP core power and 83.0% CF. The range of void fraction, axial and radial power shape, and rod positions in the core may change slightly. The effects between 120% of OLTP and 122% of OLTP are explicitly demonstrated to be inconsequential in Table 2-1 and Figures 2-1 through 2-6 supporting Methods LTR SER Limitation and Condition 9.24. While the change in power distribution in the core is achieved, the individual fuel bundles remain within the allowable thermal limits as defined in the COLR.

Also, per Methods LTR SER Limitation and Condition 9.17, the range of void fraction, axial and radial power shape, and rod positions in the core does change slightly as a result of MELLLA+ operating domain expansion including the expanded TPO region. For PBAPS, the predicted bypass void fraction at the D-Level local power range monitor (LPRM) satisfies the [[ ]] design requirement for both MELLLA+ and TPO. The steady-state bypass voiding is demonstrated on the MELLLA+ upper boundary at the TPO power level in Table 2-1.

The SRLR will validate that the power distribution in the core is achieved while maintaining individual fuel bundles within the allowable thermal limits as defined in the COLR.

As required by Methods LTR SER Limitation and Condition 9.24, the following core design and fuel monitoring parameters are plotted as indicated below in Table 2-2 and Figures 2-1 through 2-6 for each cycle exposure statepoint of the TSAR core design. The parameters are compared to the historical experience base reported in the Methods LTR (Reference 12), PBAPS PUSAR (Reference 14) and PBAPS M+SAR (Reference 15):

Table 2-2 Peak Nodal Exposures

Figure 2-1 Power of Peak Bundle versus Cycle Exposure

Figure 2-2 Coolant Flow for Peak Bundle versus Cycle Exposure

Figure 2-3 Exit Void Fraction for Peak Power Bundle versus Cycle Exposure

Figure 2-4 Maximum Channel Exit Void Fraction versus Cycle Exposure

Figure 2-5 Core Average Exit Void Fraction versus Cycle Exposure

Figure 2-6 Peak LHGR versus Cycle Exposure

In accordance with M+LTR SER Limitation and Condition 12.24.2, the exit void fraction for peak power bundle versus cycle exposure is provided in Figure 2-3.

Also, quarter core maps with mirror symmetry are plotted in Figure 2-7 through Figure 2-15 showing bundle power, bundle operating LHGR, and MCPR for beginning-of-cycle (BOC) (0 MWd/ST), middle-of-cycle (MOC) (8,000 MWd/ST), and end-of-rated (EOR) (15,850 MWd/ST) conditions. The maximum fraction of limiting power density (MFLPD) occurs at 0 MWd/ST (Figure 2-10) and the largest maximum fraction of limiting critical power ratio (MFLCPR) occurs at 10,800 MWd/ST (Figure 2-16) for this core design. In Figure 2-7 through Figure 2-9, the bundle power is dimensionless. To obtain the bundle power in MWt, multiply each number by a factor of 5.26. This factor equals 4,018/764, where 4,018 MWt is the TPO bounding thermal power and 764 is the total number of fuel bundles in the core.

Table 2-2 shows that PBAPS TPO peak nodal exposure is consistent with the PBAPS PUSAR and PBAPS M+SAR results. Figure 2-1, Figure 2-2, and Figure 2-6 show that the PBAPS TPO operation is in the expected range as compared to the reference plants and relative to PBAPS EPU and PBAPS MELLLA+. Figures 2-3 through 2-5 show that exit voiding at PBAPS is higher than other plants. The higher exit voiding is because of operating a high power density plant at lower CFs through the entire cycle. Figures 2-7 through 2-9 show the relative bundle power for BOC, MOC, and EOR, respectively. Figures 2-10 through 2-12 show the operating LHGR for BOC, MOC, and EOR, respectively. Figures 2-13 through 2-15 show the MCPR for BOC, MOC, and EOR, respectively. Figures 2-7 through 2-16 show that the general operational conditions for PBAPS in the MELLLA+ operating domain (including the expanded TPO region) are within expected parameters.

2.1.3 Fuel Thermal Monitoring Threshold

The historical 25% of LTP value for the technical specification (TS) SL, thermal limits monitoring limiting condition for operation (LCO) thresholds, and surveillance requirement (SR) thresholds are based on [[

]] The historical 25% of LTP value is a conservative basis, as described in the plant TS; [[

]]

For PBAPS EPU and MELLLA+ (120% of OLTP), the historical 25% of LTP value was reduced to 23% of LTP. [[

]] Therefore, the SL percent LTP basis, thermal limits monitoring LCOs, and SR percent LTP thresholds are changed to 22.6% of LTP for the TPO uprate.

2.2 THERMAL LIMITS ASSESSMENT

Operating thermal limits ensure that regulatory and/or safety limits are not exceeded for a range of postulated events (e.g., transients, LOCA). This section addresses the effects of TPO on thermal limits. Cycle-specific core configurations, which are evaluated for each reload, confirm TPO RTP capability and establish or confirm cycle-specific limits.

As required by Methods LTR SER Limitation and Condition 9.6, the GNF2 bundle R-factors generated for this project are consistent with GNF standard design practices, which use an axial void profile shape with 60% average in-channel voids. This is consistent with lattice axial void conditions expected for the hot channel operating state as shown in Figure 2-17.

As required by Methods LTR SER Limitation and Condition 9.15, the nodal void reactivity biases applied in TRACG are applicable to the lattices representative of fuel loaded in the core.

2.2.1 Safety Limit MCPR

The safety limit minimum critical power ratio (SLMCPR) is dependent upon the LTP and the uncertainty in its measurement. Consistent with approved practice, a SLMCPR is calculated on a cycle-specific basis for every reload using the actual core loading pattern for each reload core. The historical uncertainty allowance and calculational methods are not changed by the TPO. NRC-approved methods are used by the fuel vendor for reload licensing analysis. In the event that the cycle-specific SLMCPR is not bounded by the current PBAPS TS value, PBAPS must implement a license amendment to change the TS.

The cycle-specific SLMCPR will be determined using the methods defined in GESTAR II (Reference 16). As required by M+LTR SER Limitation and Condition 12.6, the SLMCPR will be calculated at the rated statepoint (100.0% of TPO RTP / 100.0% of CF), the upper left corner of the MELLLA+ upper boundary (100% of TPO RTP / 85.2% of CF), the lower left corner of the MELLLA+ upper boundary (77.5% of TPO RTP / 55.0% of CF), and the TPO RTP at the ICF statepoint (100.0% of TPO RTP / 110.0% of CF) (i.e., Figure 1-1 Statepoints E, J, K and F, respectively).

The currently approved off-rated CF uncertainty applied to SLO is used for the minimum CF Statepoint J and at 55.0% of CF Statepoint K. The calculated values will be documented in the SRLR. Although Statepoint E is within the MELLLA+ domain, nominal CF uncertainties are applied between 100% of CF Statepoint E and 110% of CF Statepoint F. Consistent with current practice, the CF uncertainty is applied as a CF ratio at 100% of TPO RTP when less than 100% of CF Statepoint E until application of the maximum approved off-rated CF uncertainty (or plant-specific if approved) at the minimum CF Statepoint J.

As required by Methods LTR SER Limitation and Condition 9.5, the cycle-specific SLMCPR determined based on M+LTR SER Limitation and Condition 12.6 will also include either a +0.02 SLMCPR adder for operation at statepoints with a power-to-flow ratio greater than 42 MWt/Mlbm/hr, or a +0.01 SLMCPR adder for operation at statepoints with a power-to-flow ratio less than 42 MWt/Mlbm/hr. The cycle-specific SLMCPR analysis will incorporate either a +0.01 or a +0.02 SLMCPR adder for MELLLA+ operation including the expanded TPO region.

The calculated values will be documented in the SRLR. A TS change will be requested if the current value is not bounding. The SLMCPR for PBAPS will be evaluated for the reload core prior to TPO implementation.

2.2.2 MCPR Operating Limit

The changes in core and fuel performance for a TPO [[

]]

Because the cycle-specific SLMCPR is also defined, the actual required OLMCPR can be established. This ensures an adequate fuel thermal margin for TPO uprate operation.

The power and flow dependent thermal limits are not changed with TPO as noted in TLTR SER Section 5.6.1. The sensitivity of off-rated transients to the small change in absolute power that occurs as result of retaining the same percent power for the rod block monitor (RBM) setpoints, direct scram bypass power, and thermal limits monitoring power is insignificant.

The OLMCPR is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR. The OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis, as described in GESTAR II (Reference 16). The cycle-specific analysis results are documented in the SRLR and included in the COLR. The MELLLA+ operating conditions including the expanded TPO region do not change the methods used to determine this limit.

With the usage of TRACG-AOO instead of ODYN, the +0.01 adder to the resulting OLMCPR as required by Methods LTR SER Limitation and Condition 9.19 is no longer applicable and will not be applied to the OLMCPR. The OLMCPR for PBAPS will be evaluated for the reload core prior to TPO implementation.

2.2.3 MAPLHGR Limits

The maximum average planar linear heat generation rate (MAPLHGR) is fuel dependent and not affected by PBAPS TPO operation. The ECCS performance is addressed in Section 4.3. The TPO operating conditions do not change the methods used to determine this limit.

The reload design process for PBAPS ensures that the MAPLHGR limits will be met for each reload. The MAPLHGR limits ensure that the plant does not exceed regulatory limits established in 10 CFR 50.46. Section 4.3 presents the evaluation to demonstrate that plants meet the regulatory limits in the MELLLA+ operating domain, including the expanded TPO region. [[

]] The MAPLHGR limits for PBAPS will be evaluated for the reload core prior to TPO implementation.

2.2.4 LHGR Limits

The maximum LHGR is fuel dependent and not affected by PBAPS TPO operation. The ECCS performance is addressed in Section 4.3. The TPO operating conditions do not change the methods used to determine this limit.

The maximum LHGR limits ensure that the plant does not exceed fuel thermal-mechanical DLs. The LHGR is determined by the fuel rod thermal-mechanical design and is not affected by MELLLA+ operating domain expansion from EPU or the MELLLA+ domain expansion with TPO. No changes to the fuel rod are required as a part of MELLLA+ domain expansion with TPO.

The PBAPS LHGR limits ensure that the plant does not exceed fuel thermal-mechanical DLs. There are no changes to the PBAPS fuel or fuel DLs as a result of MELLLA+ domain expansion with TPO. PBAPS continues to use the GNF2 fuel product line consistent with the GNF2 requirements (Reference 21). [[

]] The TPO operating conditions do not change the methods used to determine this limit. The LHGR limits for PBAPS will be evaluated for the reload core prior to TPO implementation.

2.2.5 Power-to-Flow Ratio

Methods LTR SER Limitation and Condition 9.3 requires that plant-specific EPU and expanded operating domain applications confirm that the core thermal power to CF ratio will not exceed 50 MWt/Mlbm/hr at any state point in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the license amendment request (LAR) will include a power distribution assessment to establish that axial and nodal power distribution uncertainties determined via neutronic methods have not increased.

The core thermal power to CF ratio at steady-state and off-rated conditions along the MELLLA+ boundary is reported in Table 2-3 and identifies the power-to-flow ratio of 50 MWt/Mlbm/hr at 55% of CF is exceeded. This condition is unchanged between MELLLA+ operation and TPO operation. The extended domain is always less than a power-to-flow ratio of 50 MWt/Mlbm/hr.

[[

]]

2.3 Reactivity Characteristics

All minimum shutdown margin requirements apply to cold shutdown conditions and are maintained without change. Checks of cold shutdown margin based on standby liquid control system (SLCS) boron injection capability and shutdown using control rods with the most reactive control rod stuck out are made for each reload. The TPO uprate has no significant effect on these conditions; the shutdown margin is confirmed in the reload core design.

The MELLLA+ operating conditions, including the expanded TPO region, do not change the PBAPS methods used to evaluate that the strongest rod out shutdown margin meets the current PBAPS design and TS cold shutdown margin requirements. The MELLLA+ operating conditions, including the expanded TPO region, do not change the PBAPS methods used to evaluate that SLCS shutdown margin meets the current PBAPS design and SLCS TS requirements.

Operation at the TPO RTP could result in a minor decrease in the hot excess reactivity during the cycle. This loss of reactivity does not affect safety and does not affect the ability to manage the power distribution through the cycle to achieve the target power level. However, the lower hot excess reactivity can result in achieving an earlier all-rods-out condition. The total cycle energy desired can be achieved through additional thermal power coastdown. Through fuel cycle redesign, sufficient excess reactivity can be obtained to match the desired cycle length. The MELLLA+ operating conditions, including the expanded TPO region, do not change the PBAPS methods used to evaluate that sufficient hot excess reactivity exists to match the 24-month cycle conditions.

2.4 THERMAL HYDRAULIC STABILITY

2.4.1 Detect and Suppress Solution – Confirmation Density

PBAPS is operating under the requirements of the stability LTS DSS-CD solution (Reference 15) consistent with the DSS-CD LTR (Reference 8), including any limitations and conditions in the applicable DSS-CD LTR SER (Reference 8). The DSS-CD stability solution has been shown to provide an early trip signal upon instability inception for both core wide and regional mode oscillations.

The DSS-CD solution monitors oscillation power range monitor (OPRM) signals to determine when a reactor scram is required. The OPRM signal is evaluated by the DSS-CD stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation (Reference 8).

[[

]]

As further discussed in Section 2.4.1 of the PBAPS M+SAR (Reference 15), Methods LTR SER Limitation and Condition 9.18 for the stability setpoints is not applicable to DSS-CD [[

]] This conclusion still applies at TPO RTP conditions.

PBAPS MELLLA+ [[ ]] stability evaluations comply with M+LTR SER Limitation and Condition 12.3.f. PBAPS [[ ]] stability evaluations at TPO RTP conditions also comply with M+LTR SER Limitation and Condition 12.3.f. The plant-specific application demonstrates that the analyses and evaluations supporting DSS-CD are applicable to the fuel loaded in the core and the new operating power [[

]]

2.4.2 Thermal Limits Monitoring Threshold

For PBAPS, the thermal limits monitoring threshold is 23.0% of EPU. For a power-uprated plant, the thermal limits monitoring threshold may be scaled to a lower percent value to maintain the same MWt. The 23.0% of CLTP boundary changes by the following equation:

TPO Thermal Limits Monitoring Threshold = 23.0% CLTP ÷ 100% TPO (% CLTP)

Thus, for a 101.66% of CLTP TPO:

TPO Thermal Limits Monitoring Threshold = 23.0% CLTP ÷ 101.66% CLTP = 22.6% TPO.

2.4.3 Armed Region

The OPRM system may only cause a scram when plant operation is in the Armed Region. In accordance with the DSS-CD LTR, the OPRM Armed Region is generically defined as the region on the P/F map at the thermal limits monitoring threshold of 25% of OLTP (23% of EPU) and rated recirculation drive flow ≤ 75% (Reference 8). For a power-uprated plant, the thermal limits monitoring threshold is scaled to a lower percent value. The rescaled thermal limits monitoring threshold becomes the new power boundary for the OPRM Armed Region boundary. For PBAPS, at TPO conditions, the new OPRM Armed Region power boundary is 22.6% of TPO.

Because the rated CF does not change for TPO, the 75% CF boundary is not rescaled.

The OPRM Armed Region for PBAPS TPO is defined as the region on the P/F map with power ≥ 22.6% of TPO RTP and a rated recirculation drive flow ≤ 75%. The OPRM Armed Region for PBAPS is illustrated in Figure 2-18.

The minimum power level at which the OPRM should be confirmed operable is 17.6% of TPO RTP. A 5% absolute power separation (i.e., 22.6% - 17.5%) between the OPRM Armed Region power boundary and the power at which the OPRM system should be confirmed operable is deemed adequate for the DSS-CD application.

Therefore, the Armed Region is deemed acceptable for TPO operation.

2.4.4 Backup Stability Protection

Two backup stability protection (BSP) options are presented in this section and summarized in Section 7.5 of Reference 8. Both options provide adequate protection for continued operation in the unlikely event the DSS-CD licensing basis algorithm cannot be demonstrated to provide its intended SLMCPR protection. The implementation for PBAPS Units 2 and 3 of both options to MELLLA+ is described in Reference 15.

The manual BSP regions are confirmed or established on a cycle-specific basis. Implementation of DSS-CD in accordance with the DSS-CD LTR (Reference 8) requires that PBAPS Units 2 and 3 confirm that the BSP approach is adequate as a part of the reload analysis. Because PBAPS Units 2 and 3 have implemented the DSS-CD solution consistent with the requirements of the DSS-CD LTR, no further review of BSP is required.

The automated backup stability protection (ABSP) setpoints [[ ]] are confirmed or established on a cycle-specific basis. Implementation of DSS-CD in accordance with the DSS-CD LTR (Reference 8) requires that PBAPS confirm that the ABSP approach is adequate as a part of the reload analysis. Because PBAPS has implemented the DSS-CD solution consistent with the requirements of the DSS-CD LTR, no further review of the ABSP is required.

As discussed in the PBAPS M+SAR (Reference 15), Section 2.4.3, appropriate TS changes have been proposed to address the implementation of DSS-CD in compliance with M+LTR SER Limitation and Condition 12.3.g. The TPO uprate does not affect this compliance.

The application of ABSP complies with M+LTR SER Limitation and Condition 12.7.

2.5 REACTIVITY CONTROL

A plant-specific evaluation was performed for PBAPS using the evaluation approach in TLTR Section 5.6.3 and Appendix J.2.3.3.

The plant-specific evaluation specifically determined that there is no change in reactor pressure, temperature, or any other condition that could affect the performance of the control rod drives (CRDs) and CRD hydraulic systems and supporting equipment.

The CRD hydraulic system is independent of power level. The increased power level will have a small effect on control blade depletion. The TPO uprate is not expected to change the cycle lifetime (replacement frequency) of any control blade. This factor will continue to be tracked per current PBAPS standard practice. Shutdown margin capability is included in each fuel reload evaluation.

The CRD system continues to meet all performance requirements at TPO uprate conditions.

2.6 ADDITIONAL LIMITATIONS AND CONDITIONS RELATED TO REACTOR CORE AND FUEL PERFORMANCE

For that subset of limitations and conditions relating to reactor core and fuel design, which did not fit conveniently into the organizational structure of the M+LTR, the required information is presented here. The information is identified by either the Methods LTR SER (Reference 12) limitation and condition (Appendix A) or the M+LTR SER (Reference 11) limitation and condition (Appendix B) to which it relates.

2.6.1 TGBLA/PANAC Version

In developing the PBAPS equilibrium core for evaluation at TPO uprated conditions, the latest versions of TGBLA and PANAC were used. Refer to Table 1-1a for the latest versions of TGBLA and PANAC. Cycle-specific analyses will include the most recent TGBLA and PANAC versions. As required by Methods LTR SER Limitation and Condition 9.1, the most recent versions of TGBLA and PANAC are used.

2.6.2 M+LTR SER Limitation and Condition 12.24.1

M+LTR SER Limitation and Condition 12.24.1 requires that the TRACG supporting analyses use the actual flow conditions. [[

]].

2.6.3 LHGR and Exposure Qualification

Methods LTR SER Limitation and Condition 9.12 states that once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P-A must utilize the PRIME thermal-mechanical methods. The PRIME LTR was approved on January 22, 2010 (Reference 9) and implemented in GESTAR II (Reference 16) in September 2010. The PBAPS M+SAR and PBAPS TSAR are based on the GNF2 fuel product line, which has a PRIME thermal-mechanical basis. PRIME fuel parameters are used in all analyses requiring fuel performance parameters.

The thermal-mechanical evaluation performed in support of the PBAPS M+SAR and PBAPS TSAR are performed using the PRIME thermal-mechanical methodology.

2.6.4 GEXL-PLUS and Pressure Drop Database

The applicability of the GNF2 experimental GEXL-PLUS and pressure drop database is confirmed for PBAPS M+SAR and PBAPS TSAR for operation in the MELLLA+ domain, including the extended TPO region.

The Methods LTR, NEDC-33173P-A (Reference 12) and this PBAPS plant-specific application of TLTR NEDC-32938P-A (Reference 1), document all analyses supporting the conclusions in this section that the application ranges of GEH codes and methods are adequate in the MELLLA+ operating domain, including the extended TPO region. In accordance with M+LTR SER Limitation and Condition 12.1, the range of mass fluxes and P/F ratios in the GEXL-PLUS database covers the intended MELLLA+ operating domain including the extended TPO region. The database includes low flows, high qualities, and void fractions. There are no restrictions on the application of the GEXL-PLUS correlation in the MELLLA+ operating domain, including the extended TPO region.

Table 2-1 Steady-State Bypass Voiding

Statepoint on P/F Map	Core Power (% of Rated)	Core Flow (% of Rated)	Hot Channel Void Fraction in Bypass Region at Instrumentation D Level (ISCOR Node 21)
E	100.0	100.0	0.00
D	100.0	101.5	0.00
J	100.0	85.2	0.00

Table 2-2 Peak Nodal Exposures

Plant	Cycle	Peak Nodal Exposure (GWd/ST)
A	18	38.849
A	19	43.784
B	9	56.359
B	10	51.544
C	7	53.447
C	8	47.766
D	13	56.660
E	11	55.387
F	EQ - 120% of OLTP	51.174
PBAPS PUSAR	EQ - 120% of OLTP	55.578
PBAPS M+SAR	EQ - 120% of OLTP	55.564
PBAPS TSAR	EQ - 122% of OLTP	55.581

Table 2-3 Core Power to Core Flow Ratio at Steady-State and Off-Rated Conditions

Statepoint on P/F Map	Core Power MWt (% of rated)	Core Flow Mlbm/hr (% of rated)	Power-to-Flow Ratio (MWt / Mlbm/hr)
E	4,016.0[1] (100.0)	102.500 (100.0)	39.20[1]
D	4,016.0[1] (100.0)	104.038 (101.5)	38.62[1]
J	4,016.0[1] (100.0)	87.330 (85.2)	46.01[1]
K	3114.0 (77.5)	56.375 (55.0)	55.24
L	2704.1 (67.3)	56.375 (55.0)	47.97

Note:

1. Evaluations are conservatively performed at the TPO bounding thermal power (4,018 MWt).

Table 2-4 [[ ]]

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[[ ]]

Table 2-5 [[ ]]

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	]]

[[

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Table 2-6 [[ ]]

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			]]

[[

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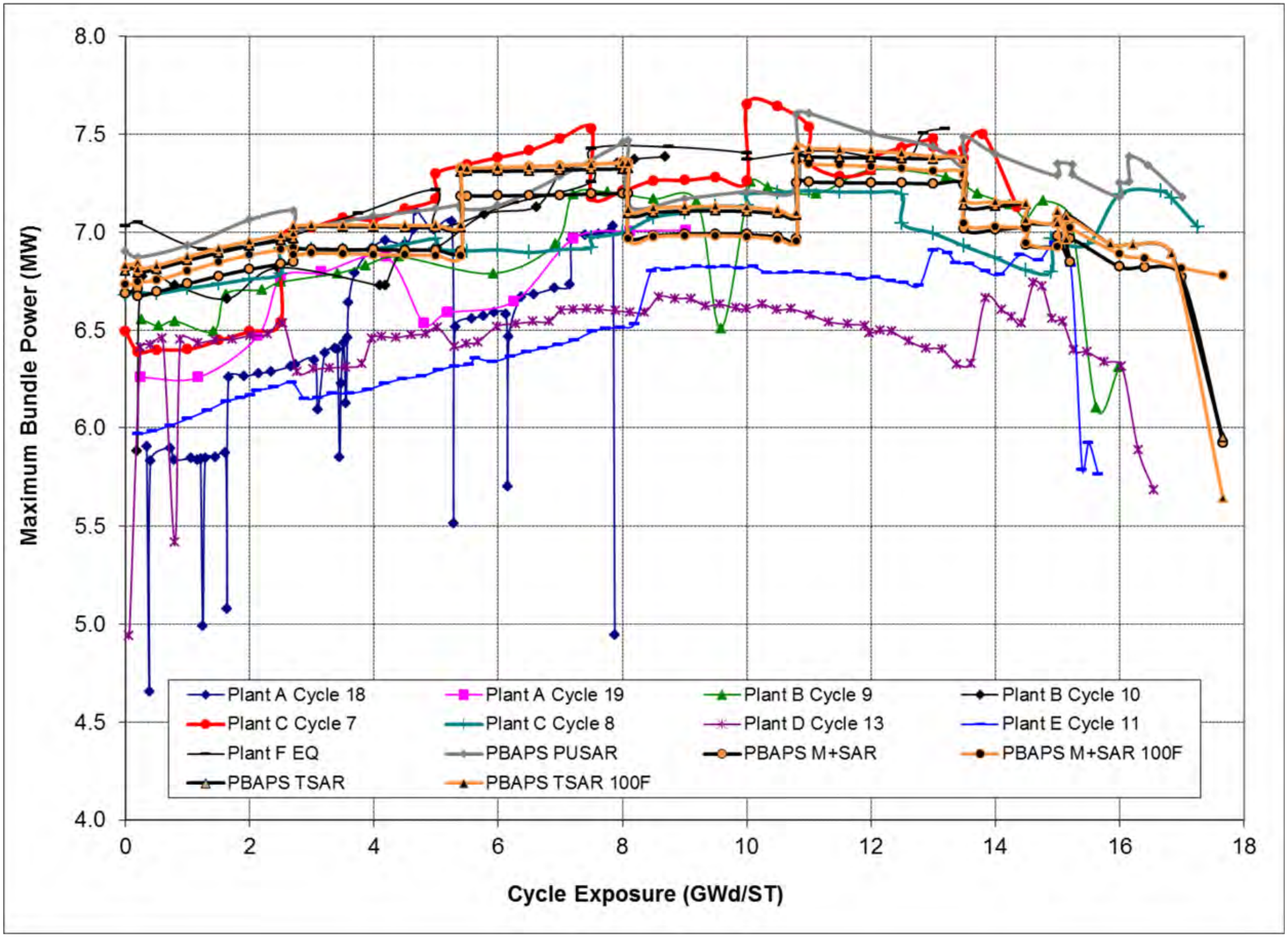


Figure 2-1 Power of Peak Bundle versus Cycle Exposure

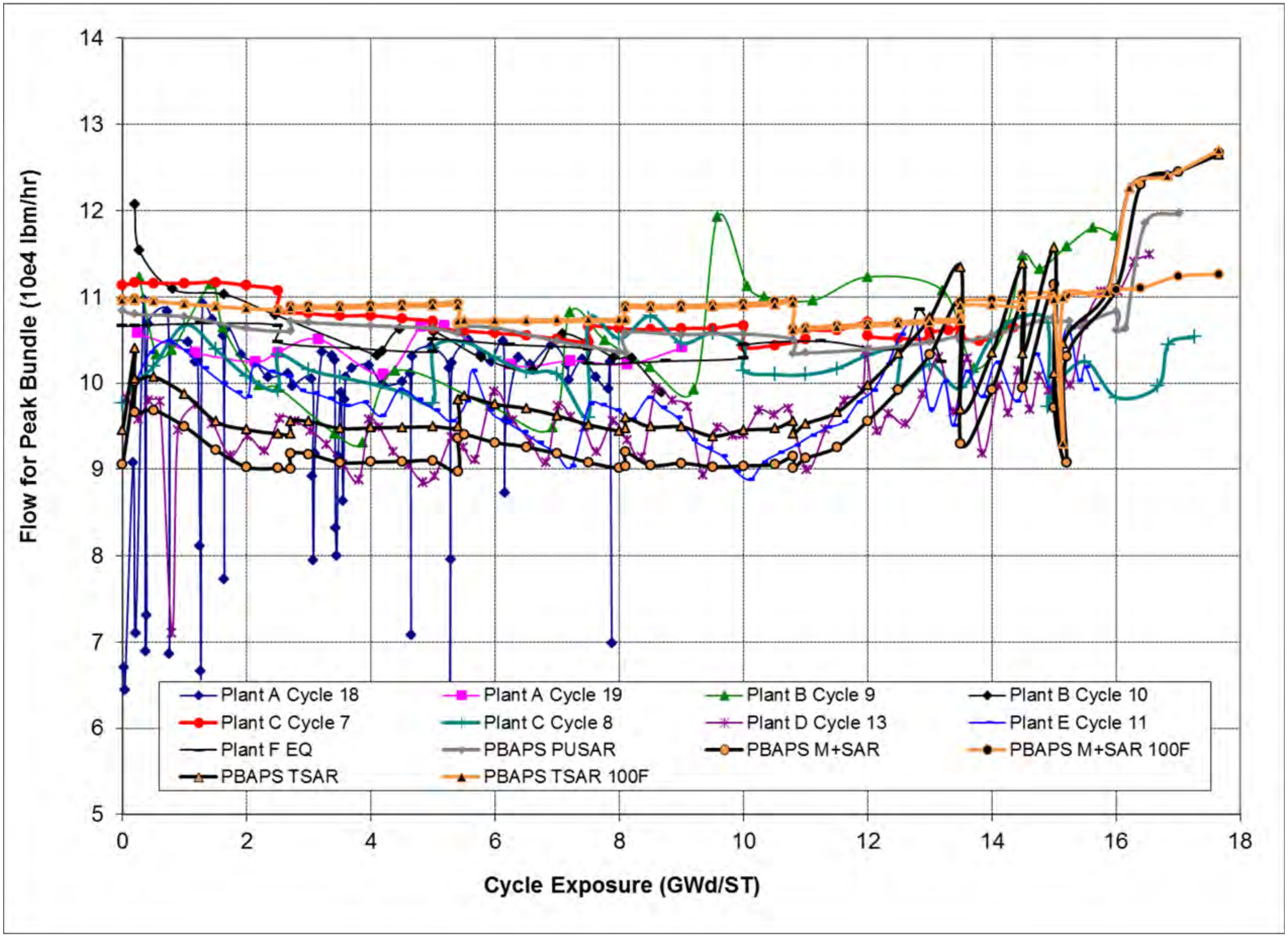


Figure 2-2 Coolant Flow for Peak Bundle versus Cycle Exposure

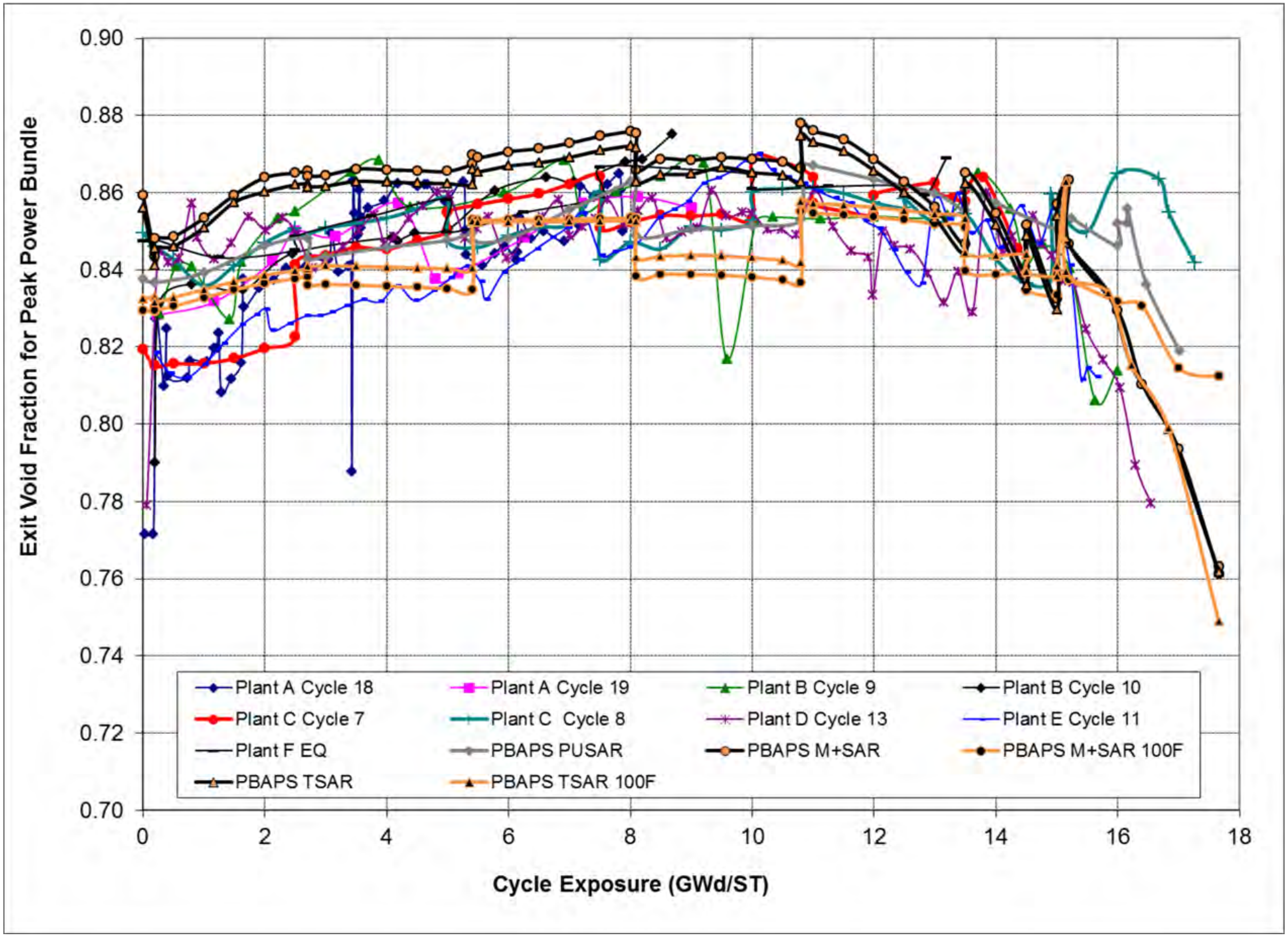


Figure 2-3 Exit Void Fraction for Peak Power Bundle versus Cycle Exposure

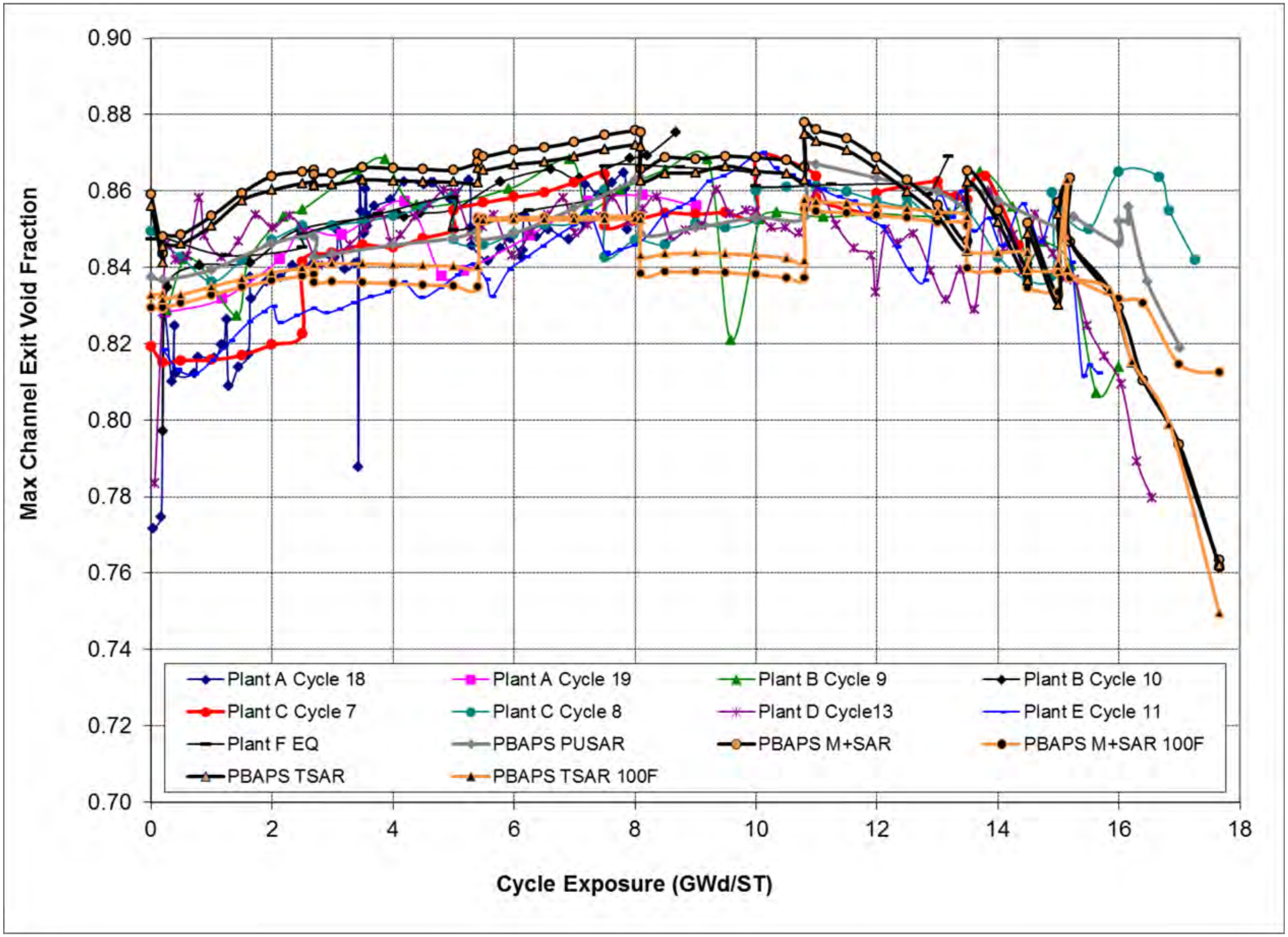


Figure 2-4 Maximum Channel Exit Void Fraction versus Cycle Exposure

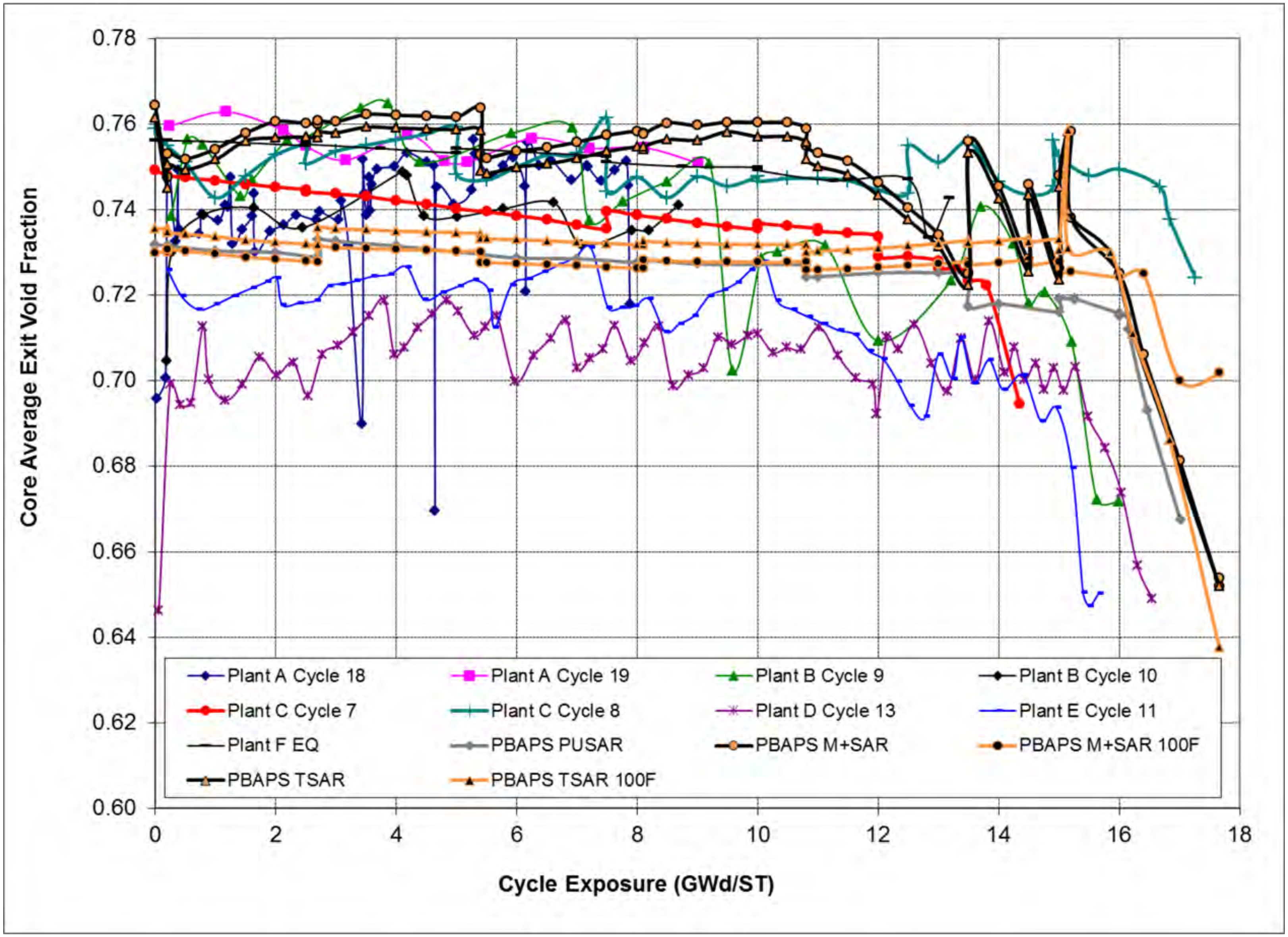


Figure 2-5 Core Average Exit Void Fraction versus Cycle Exposure

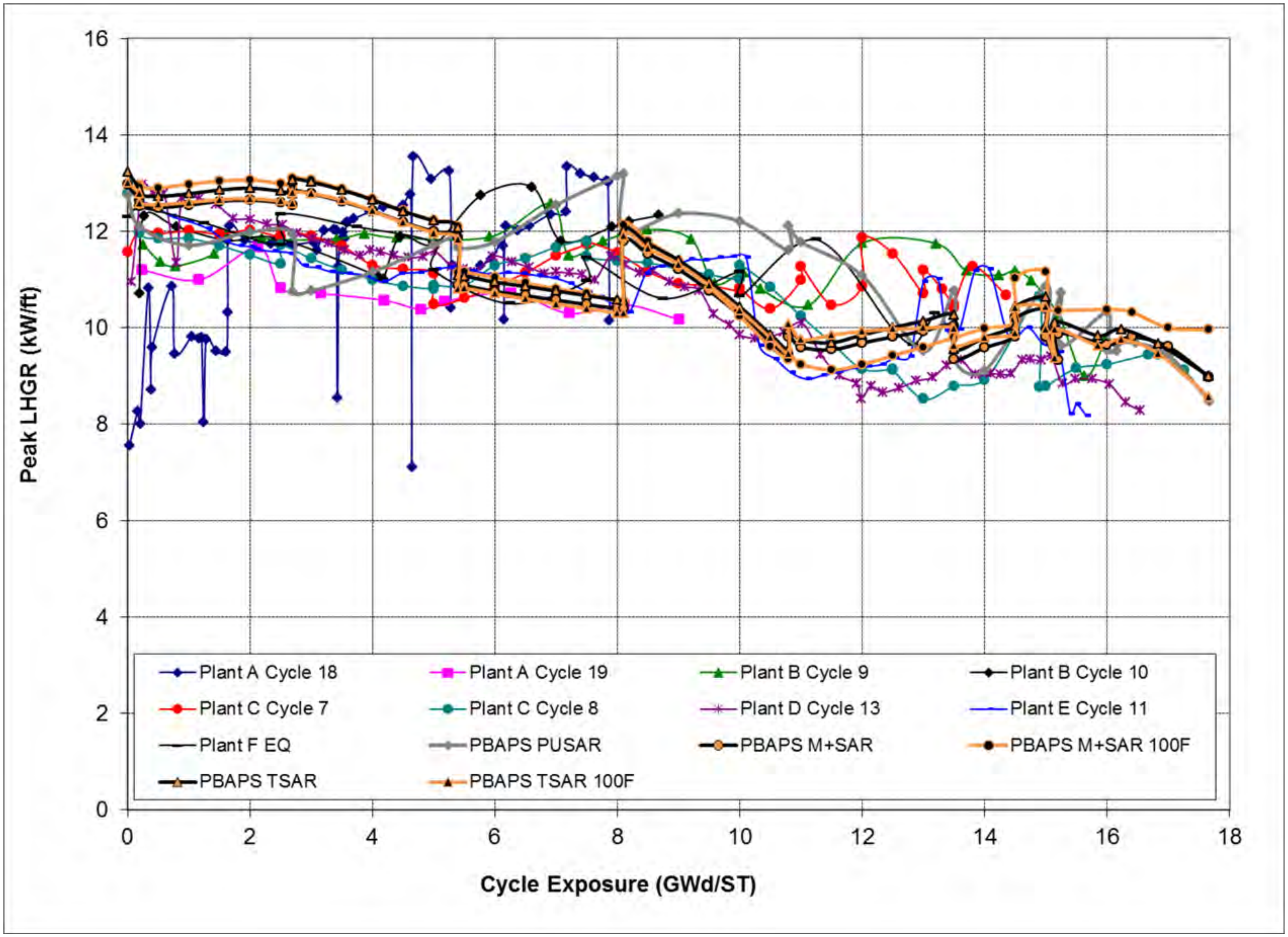
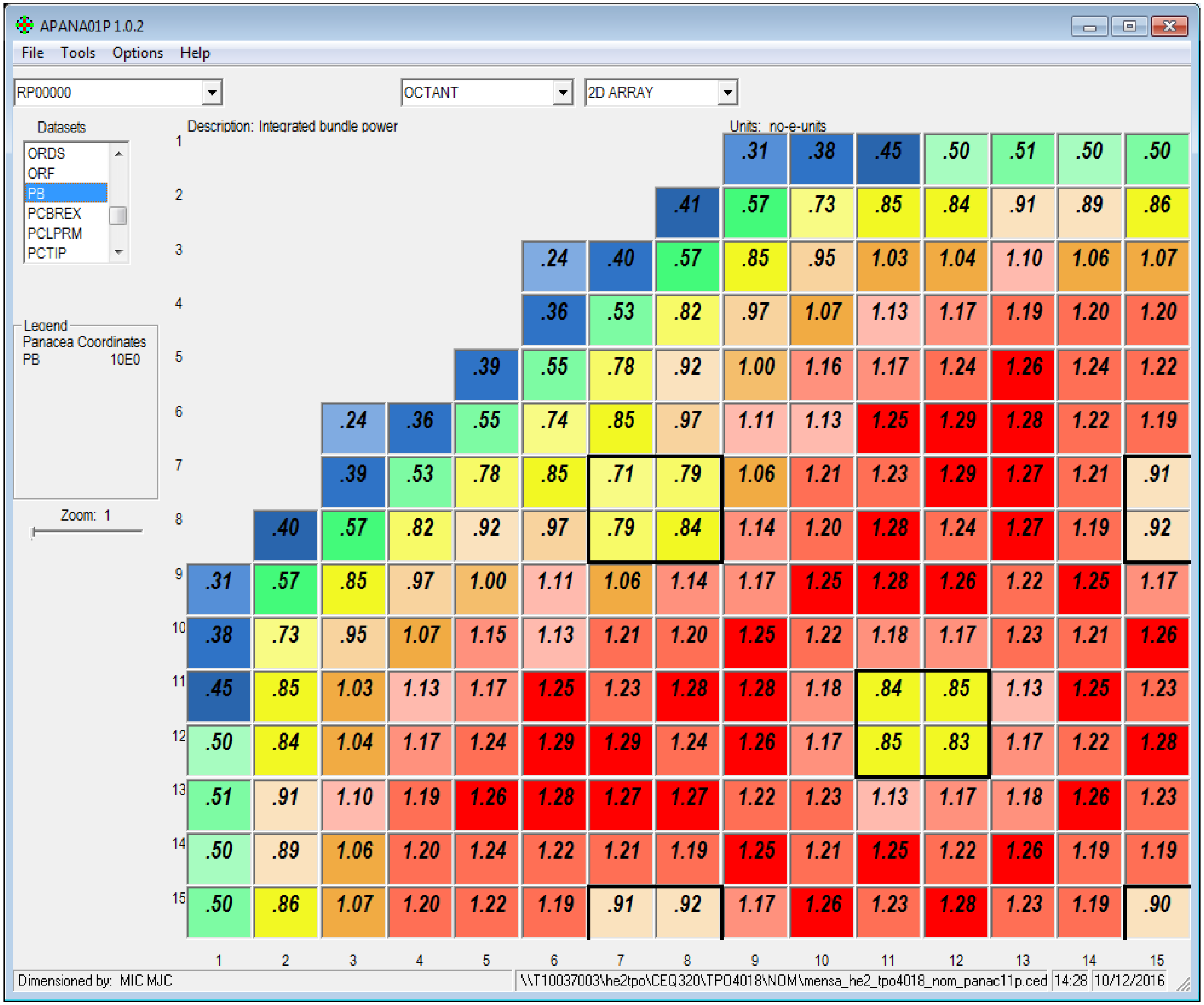
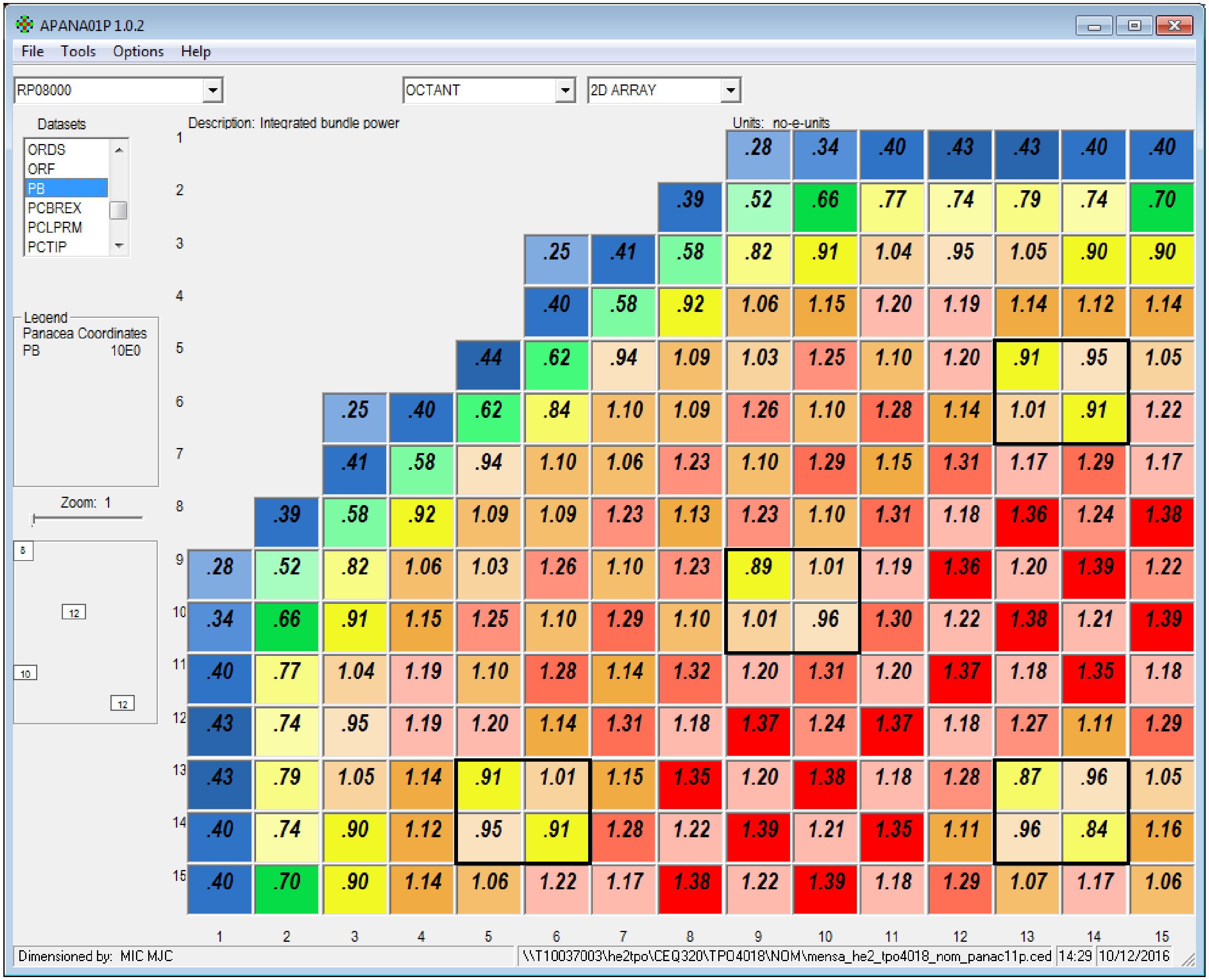


Figure 2-6 Peak LHGR versus Cycle Exposure



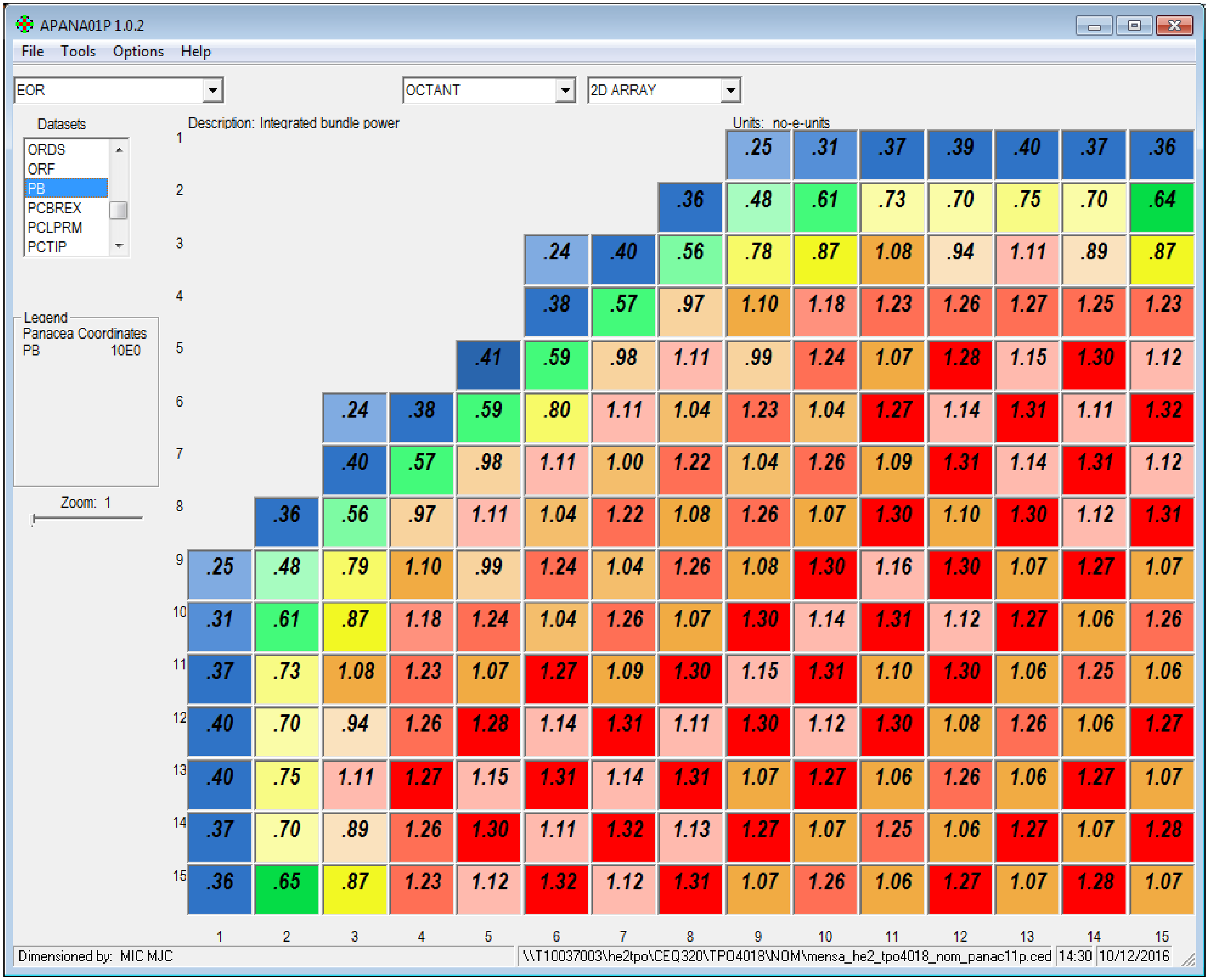
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									.31	.38	.45	.50	.51	.50	.50
2								.41	.57	.73	.85	.84	.91	.89	.86
3						.24	.40	.57	.85	.95	1.03	1.04	1.10	1.06	1.07
4						.36	.53	.82	.97	1.07	1.13	1.17	1.19	1.20	1.20
5					.39	.55	.78	.92	1.00	1.16	1.17	1.24	1.26	1.24	1.22
6			.24	.36	.55	.74	.85	.97	1.11	1.13	1.25	1.29	1.28	1.22	1.19
7			.39	.53	.78	.85	.71	.79	1.06	1.21	1.23	1.29	1.27	1.21	.91
8		.40	.57	.82	.92	.97	.79	.84	1.14	1.20	1.28	1.24	1.27	1.19	.92
9	.31	.57	.85	.97	1.00	1.11	1.06	1.14	1.17	1.25	1.28	1.26	1.22	1.25	1.17
10	.38	.73	.95	1.07	1.15	1.13	1.21	1.20	1.25	1.22	1.18	1.17	1.23	1.21	1.26
11	.45	.85	1.03	1.13	1.17	1.25	1.23	1.28	1.28	1.18	.84	.85	1.13	1.25	1.23
12	.50	.84	1.04	1.17	1.24	1.29	1.29	1.24	1.26	1.17	.85	.83	1.17	1.22	1.28
13	.51	.91	1.10	1.19	1.26	1.28	1.27	1.27	1.22	1.23	1.13	1.17	1.18	1.26	1.23
14	.50	.89	1.06	1.20	1.24	1.22	1.21	1.19	1.25	1.21	1.25	1.22	1.26	1.19	1.19
15	.50	.86	1.07	1.20	1.22	1.19	.91	.92	1.17	1.26	1.23	1.28	1.23	1.19	.90

Figure 2-7 Dimensionless Bundle Power at BOC (0 MWd/ST)



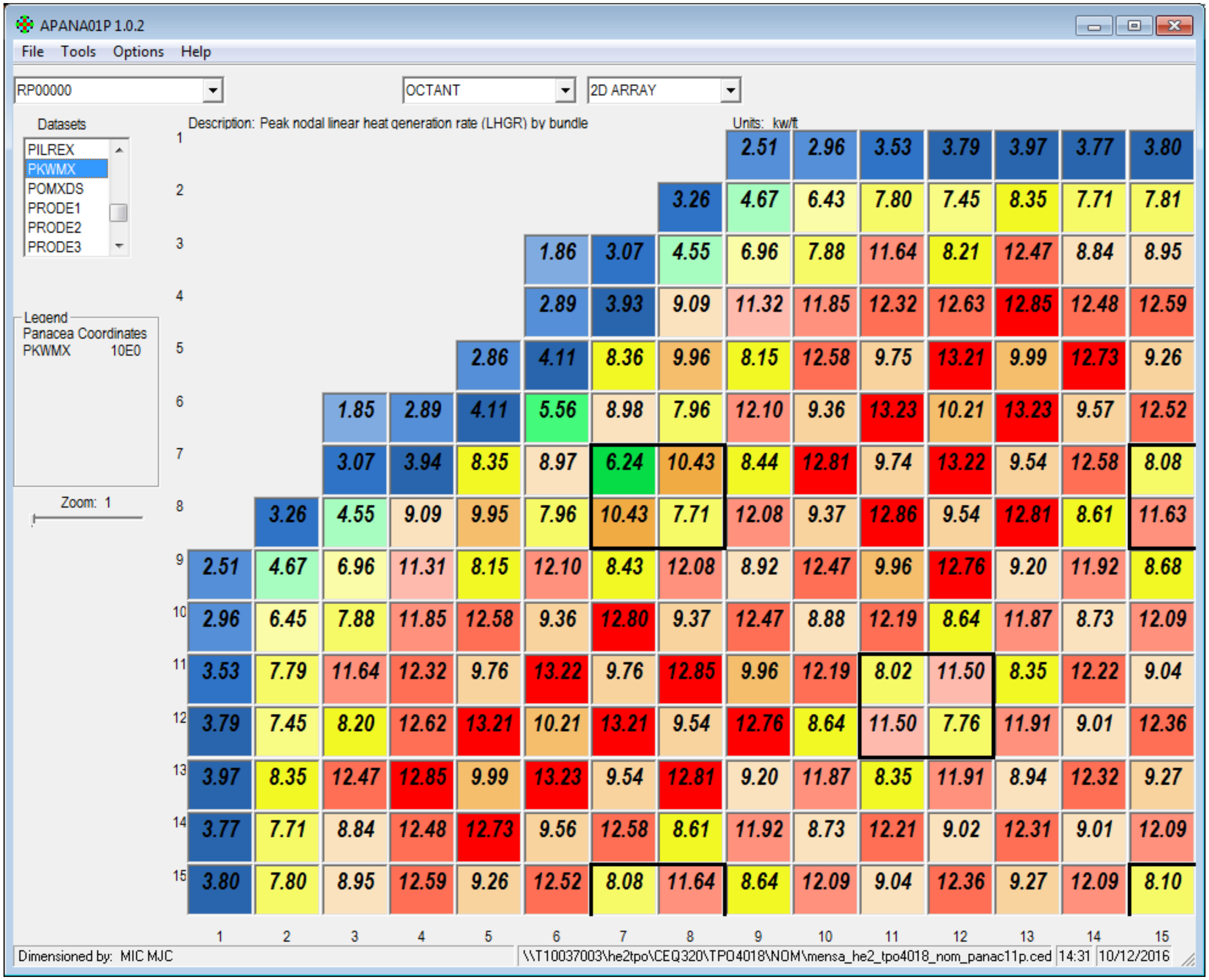
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									.28	.34	.40	.43	.43	.40	.40
2								.39	.52	.66	.77	.74	.79	.74	.70
3						.25	.41	.58	.82	.91	1.04	.95	1.05	.90	.90
4						.40	.58	.92	1.06	1.15	1.20	1.19	1.14	1.12	1.14
5					.44	.62	.94	1.09	1.03	1.25	1.10	1.20	.91	.95	1.05
6			.25	.40	.62	.84	1.10	1.09	1.26	1.10	1.28	1.14	1.01	.91	1.22
7			.41	.58	.94	1.10	1.06	1.23	1.10	1.29	1.15	1.31	1.17	1.29	1.17
8		.39	.58	.92	1.09	1.09	1.23	1.13	1.23	1.10	1.31	1.18	1.36	1.24	1.38
9	.28	.52	.82	1.06	1.03	1.26	1.10	1.23	.89	1.01	1.19	1.36	1.20	1.39	1.22
10	.34	.66	.91	1.15	1.25	1.10	1.29	1.10	1.01	.96	1.30	1.22	1.38	1.21	1.39
11	.40	.77	1.04	1.19	1.10	1.28	1.14	1.32	1.20	1.31	1.20	1.37	1.18	1.35	1.18
12	.43	.74	.95	1.19	1.20	1.14	1.31	1.18	1.37	1.24	1.37	1.18	1.27	1.11	1.29
13	.43	.79	1.05	1.14	.91	1.01	1.15	1.35	1.20	1.38	1.18	1.28	.87	.96	1.05
14	.40	.74	.90	1.12	.95	.91	1.28	1.22	1.39	1.21	1.35	1.11	.96	.84	1.16
15	.40	.70	.90	1.14	1.06	1.22	1.17	1.38	1.22	1.39	1.18	1.29	1.07	1.17	1.06

Figure 2-8 Dimensionless Bundle Power at MOC (8,000 MWd/ST)



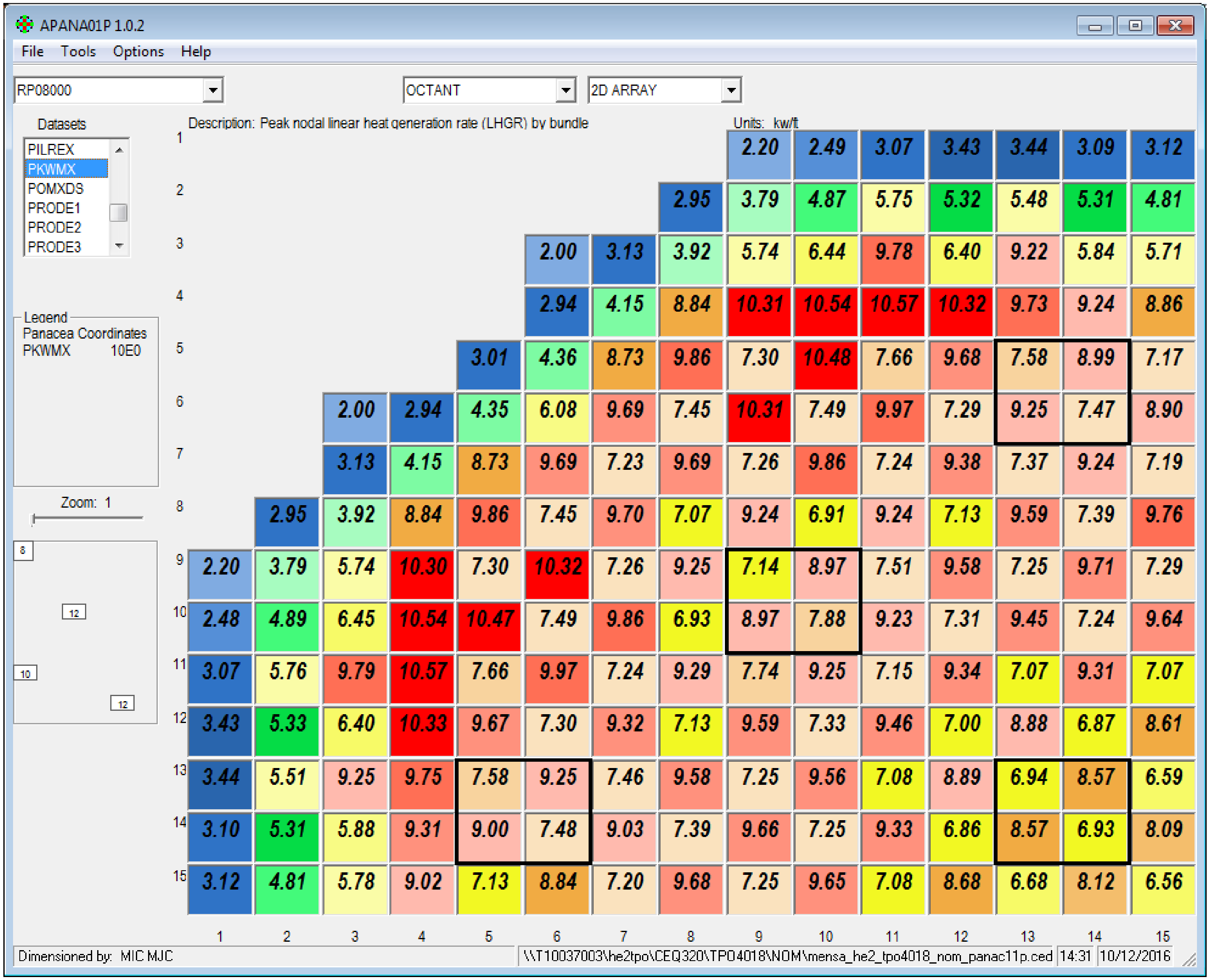
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									.25	.31	.37	.39	.40	.37	.36
2								.36	.48	.61	.73	.70	.75	.70	.64
3						.24	.40	.56	.78	.87	1.08	.94	1.11	.89	.87
4						.38	.57	.97	1.10	1.18	1.23	1.26	1.27	1.25	1.23
5					.41	.59	.98	1.11	.99	1.24	1.07	1.28	1.15	1.30	1.12
6			.24	.38	.59	.80	1.11	1.04	1.23	1.04	1.27	1.14	1.31	1.11	1.32
7			.40	.57	.98	1.11	1.00	1.22	1.04	1.26	1.09	1.31	1.14	1.31	1.12
8		.36	.56	.97	1.11	1.04	1.22	1.08	1.26	1.07	1.30	1.10	1.30	1.12	1.31
9	.25	.48	.79	1.10	.99	1.24	1.04	1.26	1.08	1.30	1.16	1.30	1.07	1.27	1.07
10	.31	.61	.87	1.18	1.24	1.04	1.26	1.07	1.30	1.14	1.31	1.12	1.27	1.06	1.26
11	.37	.73	1.08	1.23	1.07	1.27	1.09	1.30	1.15	1.31	1.10	1.30	1.06	1.25	1.06
12	.40	.70	.94	1.26	1.28	1.14	1.31	1.11	1.30	1.12	1.30	1.08	1.26	1.06	1.27
13	.40	.75	1.11	1.27	1.15	1.31	1.14	1.31	1.07	1.27	1.06	1.26	1.06	1.27	1.07
14	.37	.70	.89	1.26	1.30	1.11	1.32	1.13	1.27	1.07	1.25	1.06	1.27	1.07	1.28
15	.36	.65	.87	1.23	1.12	1.32	1.12	1.31	1.07	1.26	1.06	1.27	1.07	1.28	1.07

Figure 2-9 Dimensionless Bundle Power at EOR (15,850 MWd/ST)



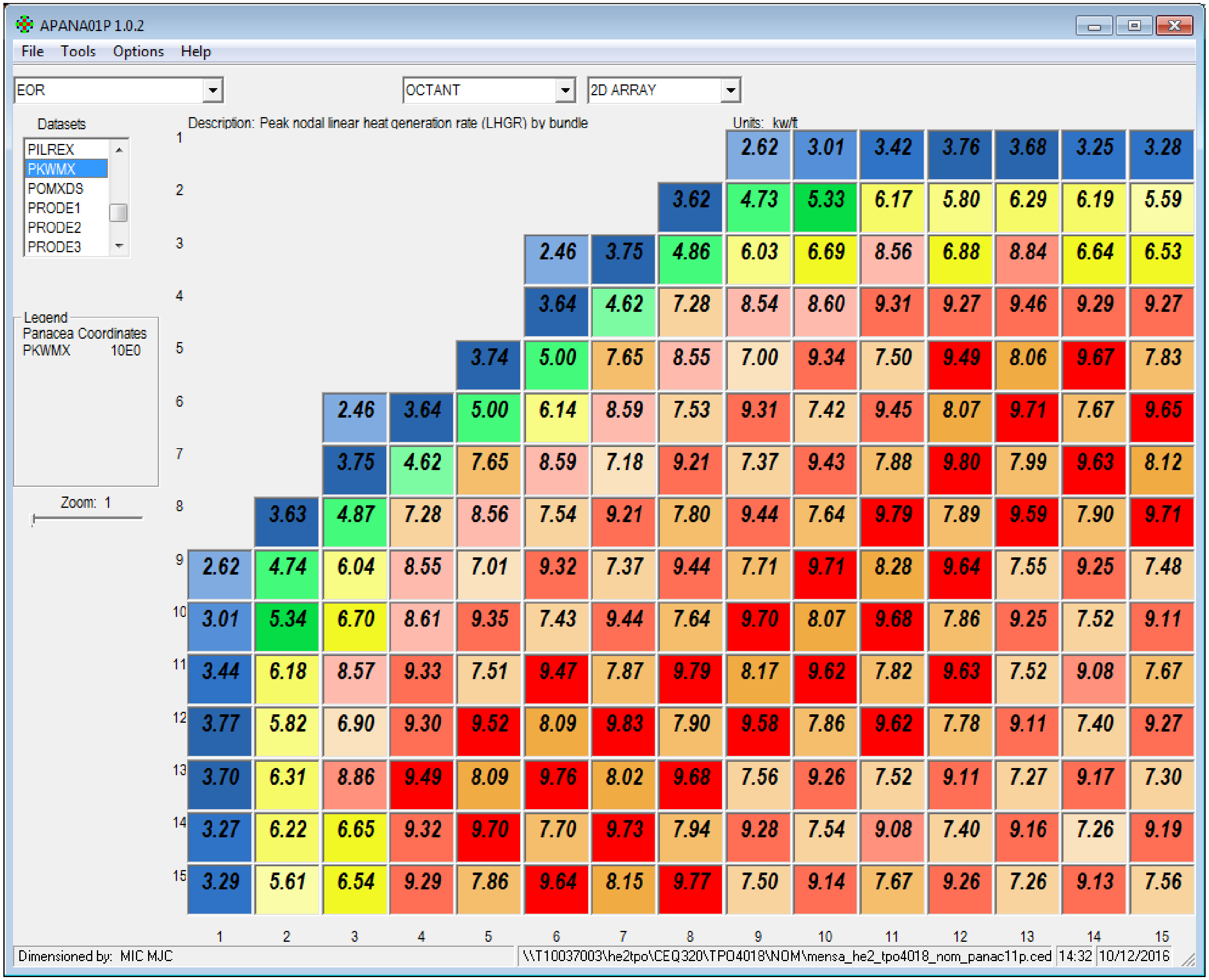
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									2.51	2.96	3.53	3.79	3.97	3.77	3.80
2								3.26	4.67	6.43	7.80	7.45	8.35	7.71	7.81
3						1.86	3.07	4.55	6.96	7.88	11.64	8.21	12.47	8.84	8.95
4						2.89	3.93	9.09	11.32	11.85	12.32	12.63	12.85	12.48	12.59
5					2.86	4.11	8.36	9.96	8.15	12.58	9.75	13.21	9.99	12.73	9.26
6			1.85	2.89	4.11	5.56	8.98	7.96	12.10	9.36	13.23	10.21	13.23	9.57	12.52
7			3.07	3.94	8.35	8.97	6.24	10.43	8.44	12.81	9.74	13.22	9.54	12.58	8.08
8		3.26	4.55	9.09	9.95	7.96	10.43	7.71	12.08	9.37	12.86	9.54	12.81	8.61	11.63
9	2.51	4.67	6.96	11.31	8.15	12.10	8.43	12.08	8.92	12.47	9.96	12.76	9.20	11.92	8.68
10	2.96	6.45	7.88	11.85	12.58	9.36	12.80	9.37	12.47	8.88	12.19	8.64	11.87	8.73	12.09
11	3.53	7.79	11.64	12.32	9.76	13.22	9.76	12.85	9.96	12.19	8.02	11.50	8.35	12.22	9.04
12	3.79	7.45	8.20	12.62	13.21	10.21	13.21	9.54	12.76	8.64	11.50	7.76	11.91	9.01	12.36
13	3.97	8.35	12.47	12.85	9.99	13.23	9.54	12.81	9.20	11.87	8.35	11.91	8.94	12.32	9.27
14	3.77	7.71	8.84	12.48	12.73	9.56	12.58	8.61	11.92	8.73	12.21	9.02	12.31	9.01	12.09
15	3.80	7.80	8.95	12.59	9.26	12.52	8.08	11.64	8.64	12.09	9.04	12.36	9.27	12.09	8.10

Figure 2-10 Bundle Operating LHGR (kW/ft) at BOC (0 MWd/ST) [Peak MFLPD Point]



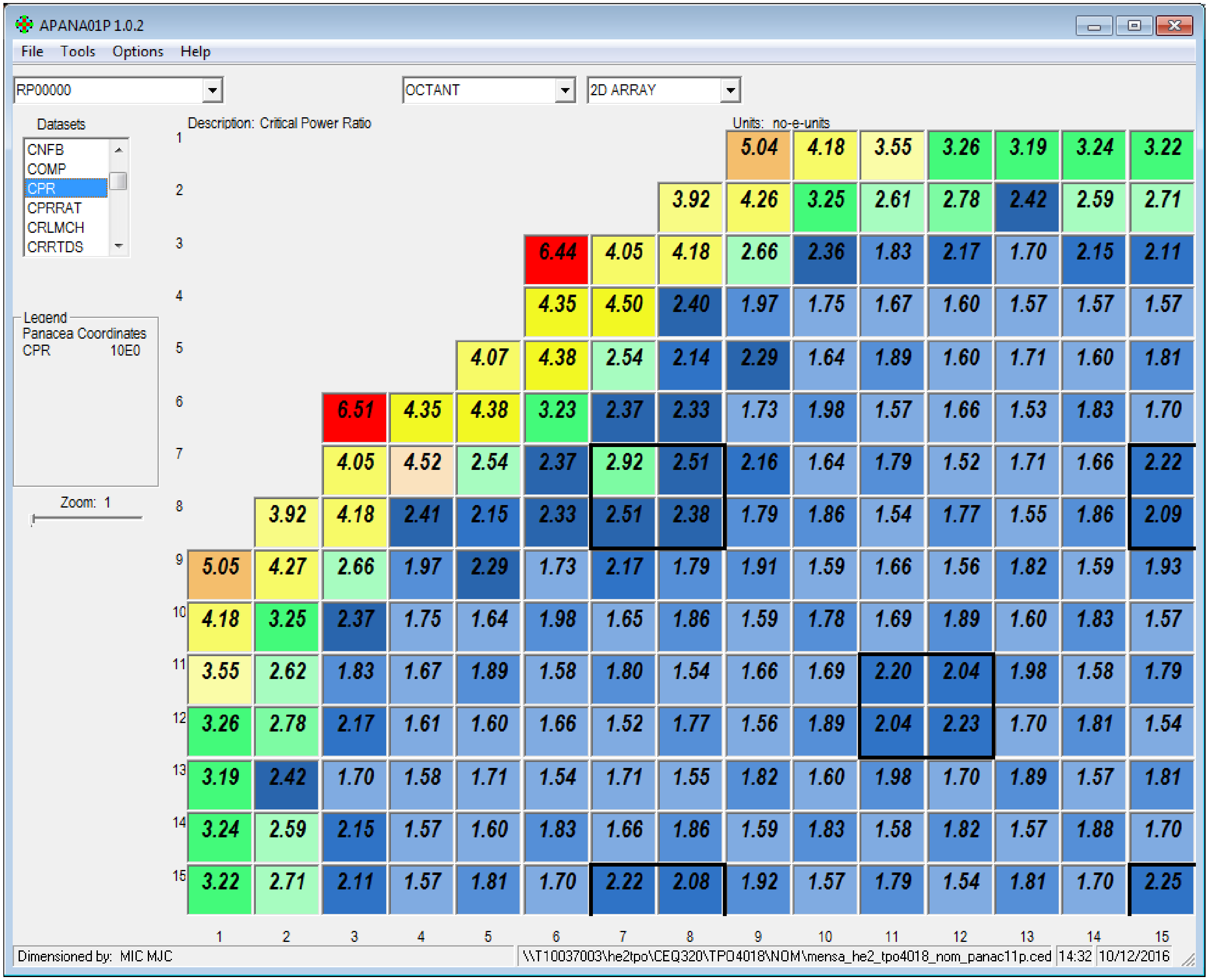
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									2.20	2.49	3.07	3.43	3.44	3.09	3.12
2								2.95	3.79	4.87	5.75	5.32	5.48	5.31	4.81
3						2.00	3.13	3.92	5.74	6.44	9.78	6.40	9.22	5.84	5.71
4						2.94	4.15	8.84	10.31	10.54	10.57	10.32	9.73	9.24	8.86
5					3.01	4.36	8.73	9.86	7.30	10.48	7.66	9.68	7.58	8.99	7.17
6			2.00	2.94	4.35	6.08	9.69	7.45	10.31	7.49	9.97	7.29	9.25	7.47	8.90
7			3.13	4.15	8.73	9.69	7.23	9.69	7.26	9.86	7.24	9.38	7.37	9.24	7.19
8		2.95	3.92	8.84	9.86	7.45	9.70	7.07	9.24	6.91	9.24	7.13	9.59	7.39	9.76
9	2.20	3.79	5.74	10.30	7.30	10.32	7.26	9.25	7.14	8.97	7.51	9.58	7.25	9.71	7.29
10	2.48	4.89	6.45	10.54	10.47	7.49	9.86	6.93	8.97	7.88	9.23	7.31	9.45	7.24	9.64
11	3.07	5.76	9.79	10.57	7.66	9.97	7.24	9.29	7.74	9.25	7.15	9.34	7.07	9.31	7.07
12	3.43	5.33	6.40	10.33	9.67	7.30	9.32	7.13	9.59	7.33	9.46	7.00	8.88	6.87	8.61
13	3.44	5.51	9.25	9.75	7.58	9.25	7.46	9.58	7.25	9.56	7.08	8.89	6.94	8.57	6.59
14	3.10	5.31	5.88	9.31	9.00	7.48	9.03	7.39	9.66	7.25	9.33	6.86	8.57	6.93	8.09
15	3.12	4.81	5.78	9.02	7.13	8.84	7.20	9.68	7.25	9.65	7.08	8.68	6.68	8.12	6.56

Figure 2-11 Bundle Operating LHGR (kW/ft) at MOC (8,000 MWd/ST)



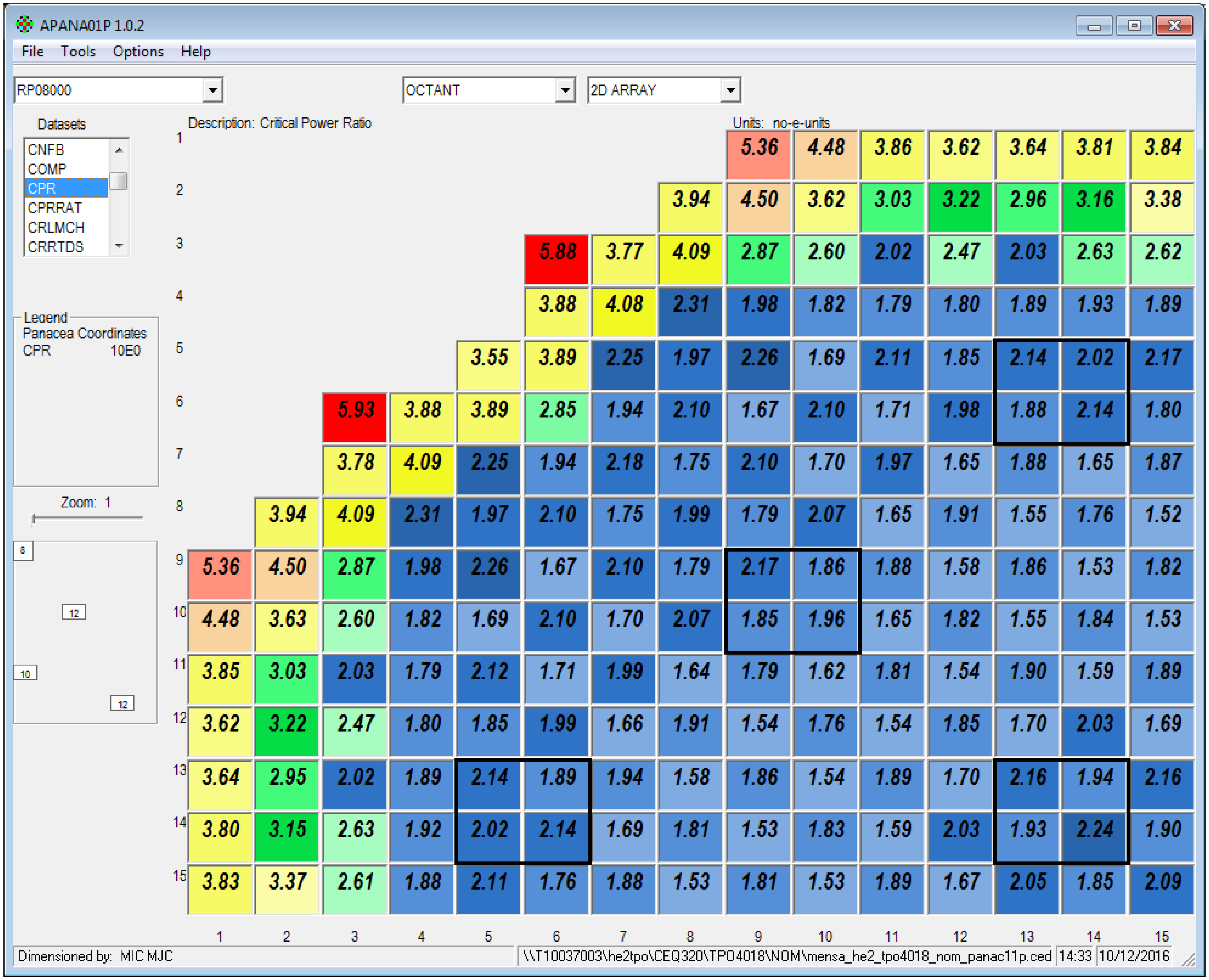
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									2.62	3.01	3.42	3.76	3.68	3.25	3.28
2								3.62	4.73	5.33	6.17	5.80	6.29	6.19	5.59
3						2.46	3.75	4.86	6.03	6.69	8.56	6.88	8.84	6.64	6.53
4						3.64	4.62	7.28	8.54	8.60	9.31	9.27	9.46	9.29	9.27
5					3.74	5.00	7.65	8.55	7.00	9.34	7.50	9.49	8.06	9.67	7.83
6			2.46	3.64	5.00	6.14	8.59	7.53	9.31	7.42	9.45	8.07	9.71	7.67	9.65
7			3.75	4.62	7.65	8.59	7.18	9.21	7.37	9.43	7.88	9.80	7.99	9.63	8.12
8		3.63	4.87	7.28	8.56	7.54	9.21	7.80	9.44	7.64	9.79	7.89	9.59	7.90	9.71
9	2.62	4.74	6.04	8.55	7.01	9.32	7.37	9.44	7.71	9.71	8.28	9.64	7.55	9.25	7.48
10	3.01	5.34	6.70	8.61	9.35	7.43	9.44	7.64	9.70	8.07	9.68	7.86	9.25	7.52	9.11
11	3.44	6.18	8.57	9.33	7.51	9.47	7.87	9.79	8.17	9.62	7.82	9.63	7.52	9.08	7.67
12	3.77	5.82	6.90	9.30	9.52	8.09	9.83	7.90	9.58	7.86	9.62	7.78	9.11	7.40	9.27
13	3.70	6.31	8.86	9.49	8.09	9.76	8.02	9.68	7.56	9.26	7.52	9.11	7.27	9.17	7.30
14	3.27	6.22	6.65	9.32	9.70	7.70	9.73	7.94	9.28	7.54	9.08	7.40	9.16	7.26	9.19
15	3.29	5.61	6.54	9.29	7.86	9.64	8.15	9.77	7.50	9.14	7.67	9.26	7.26	9.13	7.56

Figure 2-12 Bundle Operating LHGR (kW/ft) at EOR (15,850 MWd/ST)



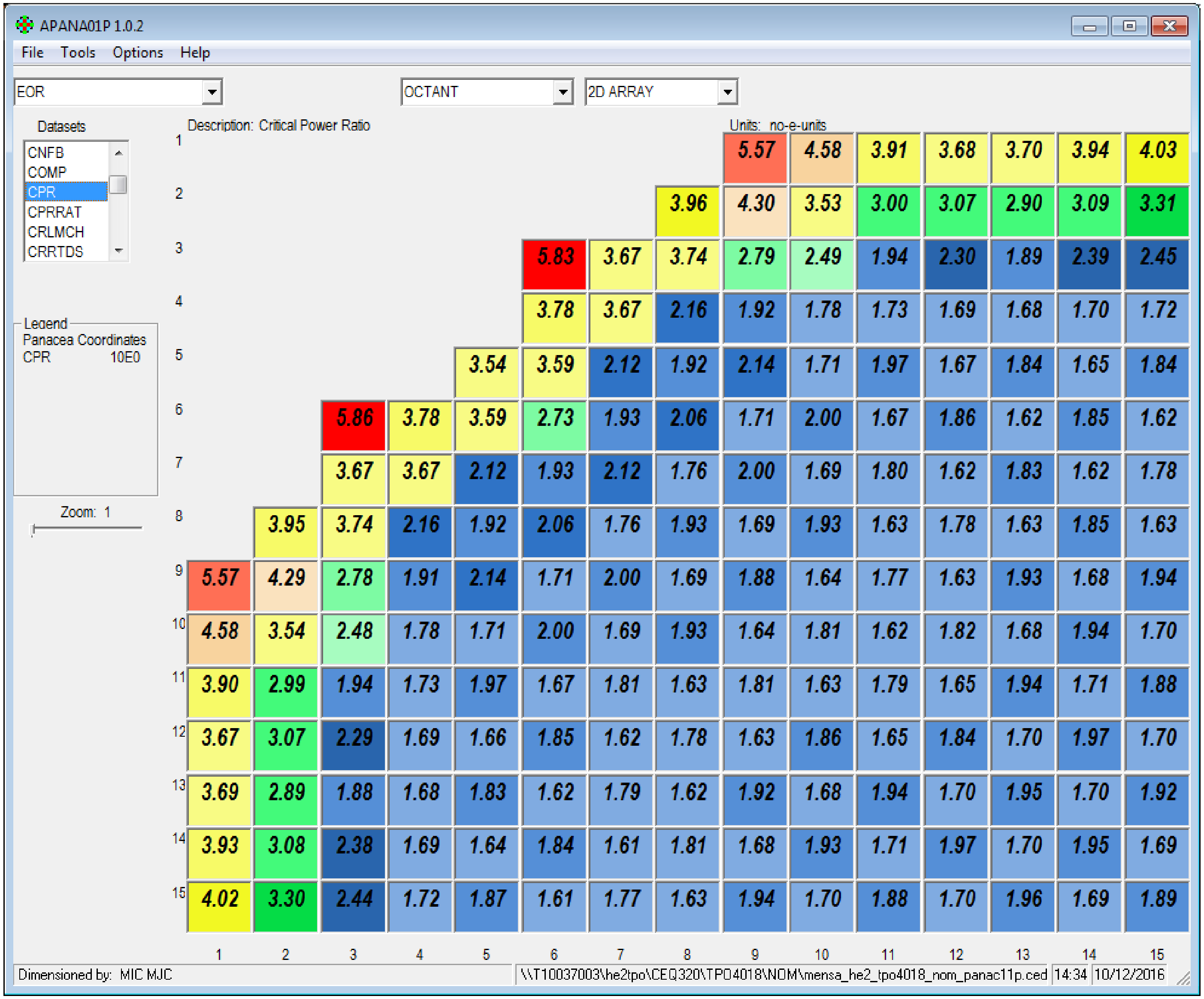
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									5.04	4.18	3.55	3.26	3.19	3.24	3.22
2								3.92	4.26	3.25	2.61	2.78	2.42	2.59	2.71
3						6.44	4.05	4.18	2.66	2.36	1.83	2.17	1.70	2.15	2.11
4						4.35	4.50	2.40	1.97	1.75	1.67	1.60	1.57	1.57	1.57
5					4.07	4.38	2.54	2.14	2.29	1.64	1.89	1.60	1.71	1.60	1.81
6			6.51	4.35	4.38	3.23	2.37	2.33	1.73	1.98	1.57	1.66	1.53	1.83	1.70
7			4.05	4.52	2.54	2.37	2.92	2.51	2.16	1.64	1.79	1.52	1.71	1.66	2.22
8		3.92	4.18	2.41	2.15	2.33	2.51	2.38	1.79	1.86	1.54	1.77	1.55	1.86	2.09
9	5.05	4.27	2.66	1.97	2.29	1.73	2.17	1.79	1.91	1.59	1.66	1.56	1.82	1.59	1.93
10	4.18	3.25	2.37	1.75	1.64	1.98	1.65	1.86	1.59	1.78	1.69	1.89	1.60	1.83	1.57
11	3.55	2.62	1.83	1.67	1.89	1.58	1.80	1.54	1.66	1.69	2.20	2.04	1.98	1.58	1.79
12	3.26	2.78	2.17	1.61	1.60	1.66	1.52	1.77	1.56	1.89	2.04	2.23	1.70	1.81	1.54
13	3.19	2.42	1.70	1.58	1.71	1.54	1.71	1.55	1.82	1.60	1.98	1.70	1.89	1.57	1.81
14	3.24	2.59	2.15	1.57	1.60	1.83	1.66	1.86	1.59	1.83	1.58	1.82	1.57	1.88	1.70
15	3.22	2.71	2.11	1.57	1.81	1.70	2.22	2.08	1.92	1.57	1.79	1.54	1.81	1.70	2.25

Figure 2-13 Bundle Operating MCPR at BOC (0 MWd/ST)



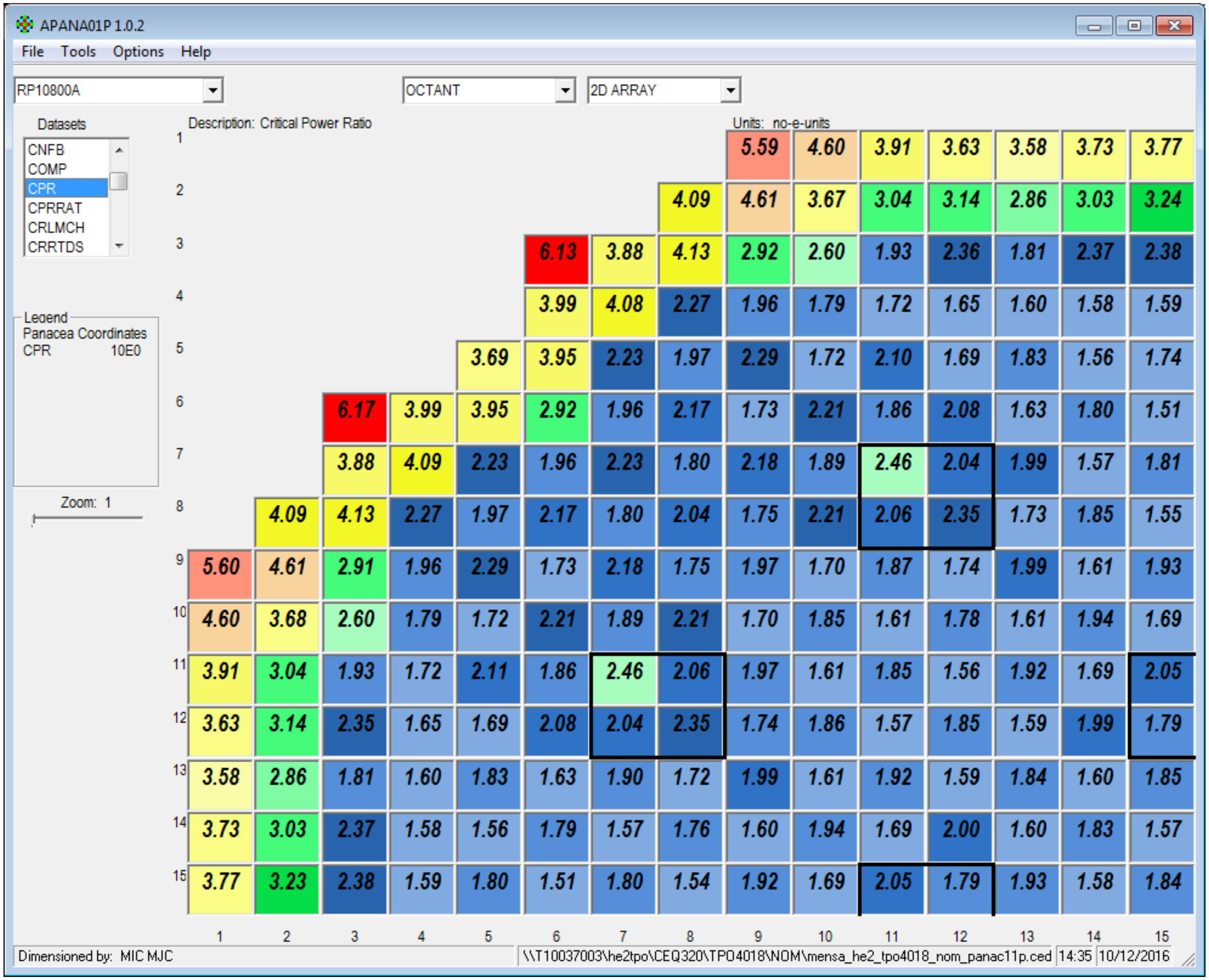
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									5.36	4.48	3.86	3.62	3.64	3.81	3.84
2								3.94	4.50	3.62	3.03	3.22	2.96	3.16	3.38
3						5.88	3.77	4.09	2.87	2.60	2.02	2.47	2.03	2.63	2.62
4						3.88	4.08	2.31	1.98	1.82	1.79	1.80	1.89	1.93	1.89
5					3.55	3.89	2.25	1.97	2.26	1.69	2.11	1.85	2.14	2.02	2.17
6			5.93	3.88	3.89	2.85	1.94	2.10	1.67	2.10	1.71	1.98	1.88	2.14	1.80
7			3.78	4.09	2.25	1.94	2.18	1.75	2.10	1.70	1.97	1.65	1.88	1.65	1.87
8		3.94	4.09	2.31	1.97	2.10	1.75	1.99	1.79	2.07	1.65	1.91	1.55	1.76	1.52
9	5.36	4.50	2.87	1.98	2.26	1.67	2.10	1.79	2.17	1.86	1.88	1.58	1.86	1.53	1.82
10	4.48	3.63	2.60	1.82	1.69	2.10	1.70	2.07	1.85	1.96	1.65	1.82	1.55	1.84	1.53
11	3.85	3.03	2.03	1.79	2.12	1.71	1.99	1.64	1.79	1.62	1.81	1.54	1.90	1.59	1.89
12	3.62	3.22	2.47	1.80	1.85	1.99	1.66	1.91	1.54	1.76	1.54	1.85	1.70	2.03	1.69
13	3.64	2.95	2.02	1.89	2.14	1.89	1.94	1.58	1.86	1.54	1.89	1.70	2.16	1.94	2.16
14	3.80	3.15	2.63	1.92	2.02	2.14	1.69	1.81	1.53	1.83	1.59	2.03	1.93	2.24	1.90
15	3.83	3.37	2.61	1.88	2.11	1.76	1.88	1.53	1.81	1.53	1.89	1.67	2.05	1.85	2.09

Figure 2-14 Bundle Operating MCPR at MOC (8,000 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									5.57	4.58	3.91	3.68	3.70	3.94	4.03
2								3.96	4.30	3.53	3.00	3.07	2.90	3.09	3.31
3						5.83	3.67	3.74	2.79	2.49	1.94	2.30	1.89	2.39	2.45
4						3.78	3.67	2.16	1.92	1.78	1.73	1.69	1.68	1.70	1.72
5					3.54	3.59	2.12	1.92	2.14	1.71	1.97	1.67	1.84	1.65	1.84
6			5.86	3.78	3.59	2.73	1.93	2.06	1.71	2.00	1.67	1.86	1.62	1.85	1.62
7			3.67	3.67	2.12	1.93	2.12	1.76	2.00	1.69	1.80	1.62	1.83	1.62	1.78
8		3.95	3.74	2.16	1.92	2.06	1.76	1.93	1.69	1.93	1.63	1.78	1.63	1.85	1.63
9	5.57	4.29	2.78	1.91	2.14	1.71	2.00	1.69	1.88	1.64	1.77	1.63	1.93	1.68	1.94
10	4.58	3.54	2.48	1.78	1.71	2.00	1.69	1.93	1.64	1.81	1.62	1.82	1.68	1.94	1.70
11	3.90	2.99	1.94	1.73	1.97	1.67	1.81	1.63	1.81	1.63	1.79	1.65	1.94	1.71	1.88
12	3.67	3.07	2.29	1.69	1.66	1.85	1.62	1.78	1.63	1.86	1.65	1.84	1.70	1.97	1.70
13	3.69	2.89	1.88	1.68	1.83	1.62	1.79	1.62	1.92	1.68	1.94	1.70	1.95	1.70	1.92
14	3.93	3.08	2.38	1.69	1.64	1.84	1.61	1.81	1.68	1.93	1.71	1.97	1.70	1.95	1.69
15	4.02	3.30	2.44	1.72	1.87	1.61	1.77	1.63	1.94	1.70	1.88	1.70	1.96	1.69	1.89

Figure 2-15 Bundle Operating MCPR at EOR (15,850 MWd/ST)



	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1									5.59	4.60	3.91	3.63	3.58	3.73	3.77
2								4.09	4.61	3.67	3.04	3.14	2.86	3.03	3.24
3						6.13	3.88	4.13	2.92	2.60	1.93	2.36	1.81	2.37	2.38
4						3.99	4.08	2.27	1.96	1.79	1.72	1.65	1.60	1.58	1.59
5					3.69	3.95	2.23	1.97	2.29	1.72	2.10	1.69	1.83	1.56	1.74
6			6.17	3.99	3.95	2.92	1.96	2.17	1.73	2.21	1.86	2.08	1.63	1.80	1.51
7			3.88	4.09	2.23	1.96	2.23	1.80	2.18	1.89	2.46	2.04	1.99	1.57	1.81
8		4.09	4.13	2.27	1.97	2.17	1.80	2.04	1.75	2.21	2.06	2.35	1.73	1.85	1.55
9	5.60	4.61	2.91	1.96	2.29	1.73	2.18	1.75	1.97	1.70	1.87	1.74	1.99	1.61	1.93
10	4.60	3.68	2.60	1.79	1.72	2.21	1.89	2.21	1.70	1.85	1.61	1.78	1.61	1.94	1.69
11	3.91	3.04	1.93	1.72	2.11	1.86	2.46	2.06	1.97	1.61	1.85	1.56	1.92	1.69	2.05
12	3.63	3.14	2.35	1.65	1.69	2.08	2.04	2.35	1.74	1.86	1.57	1.85	1.59	1.99	1.79
13	3.58	2.86	1.81	1.60	1.83	1.63	1.90	1.72	1.99	1.61	1.92	1.59	1.84	1.60	1.85
14	3.73	3.03	2.37	1.58	1.56	1.79	1.57	1.76	1.60	1.94	1.69	2.00	1.60	1.83	1.57
15	3.77	3.23	2.38	1.59	1.80	1.51	1.80	1.54	1.92	1.69	2.05	1.79	1.93	1.58	1.84

Figure 2-16 Bundle Operating MCPR at 10,800 MWd/ST
[Peak MFLCPR Point]

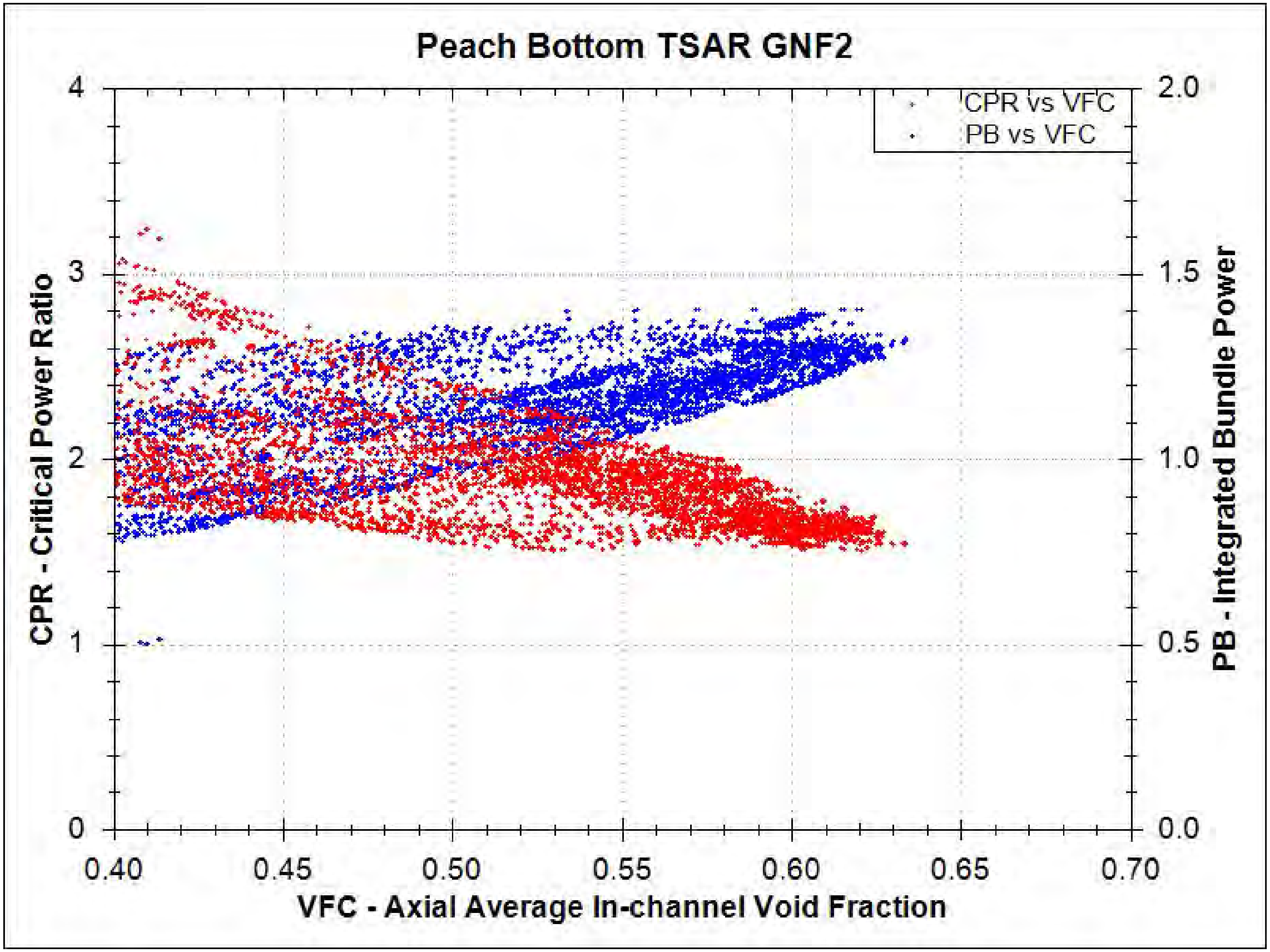


Figure 2-17 Bundle Average Void Fraction versus Critical Power and Bundle Power

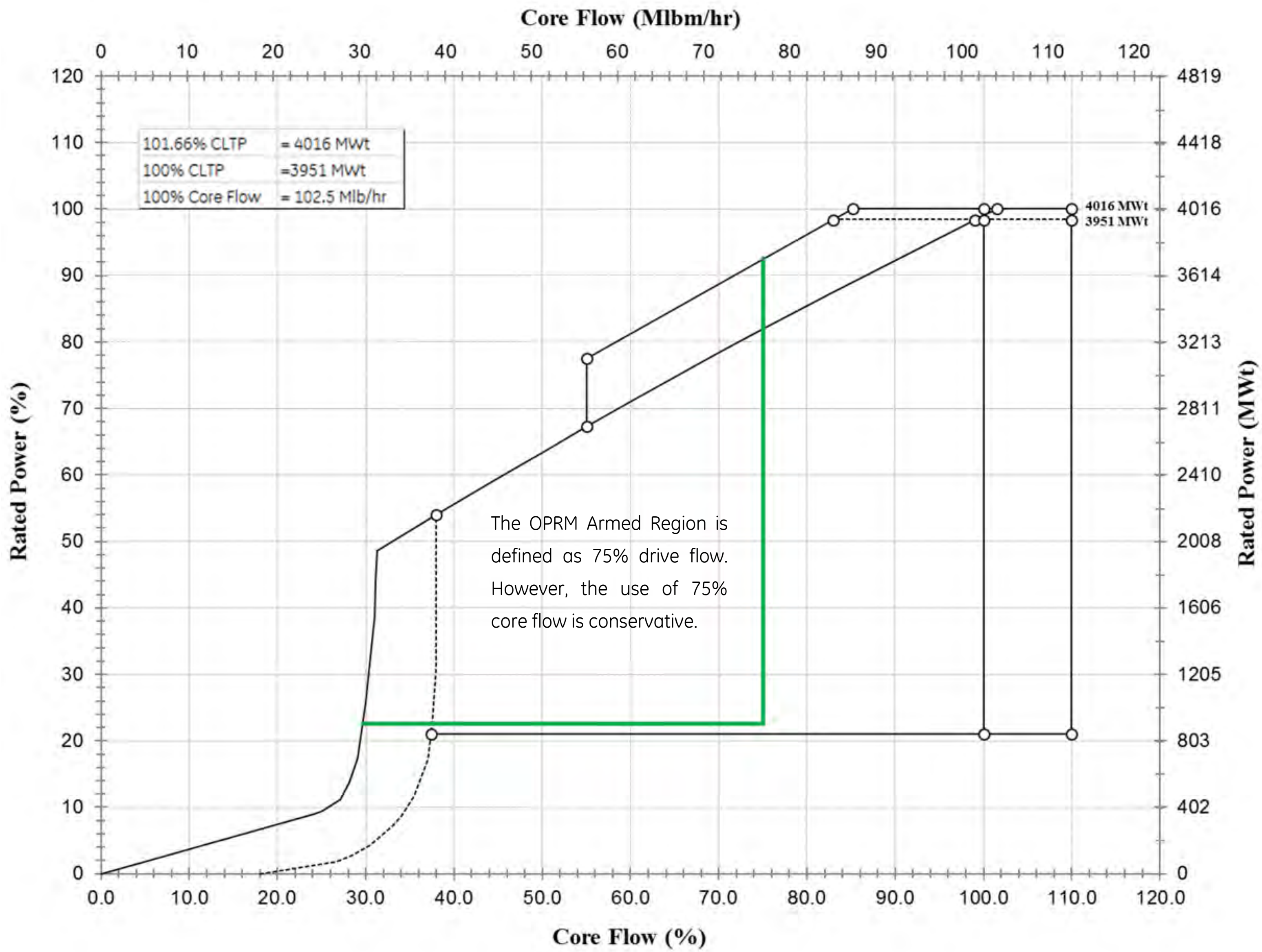


Figure 2-18 Required OPRM Armed Region

3.0 REACTOR COOLANT AND CONNECTED SYSTEMS

3.1 NUCLEAR SYSTEM PRESSURE RELIEF / OVERPRESSURE PROTECTION

The pressure relief system prevents over-pressurization of the nuclear system during abnormal operational transients. The SRVs, along with other functions, provide this protection. The EPU evaluation (Reference 14), using the approach described in the TLTR (Reference 1), Section 5.6.8, was previously performed at 102% of CLTP to demonstrate that the reactor vessel conformed to ASME Boiler and Pressure Vessel (B&PV) Code and plant TS requirements.

A plant-specific evaluation of the effects of TPO RTP compared to CLTP determined that:

- There is no increase in nominal operating pressure for the PBAPS TPO uprate;
- There are no changes in the SRV setpoints or valve OOS options; and
- There is no change in the methodology or the limiting overpressure event.

It is concluded that:

- Because the current ASME overpressure analysis accounts for ≥102% of CLTP and the ASME overpressure analysis for the first TPO uprate cycle will also account for ≥102% of CLTP, the relief capacity of the SRVs is not affected by the TPO uprate;
- The first TPO uprate cycle reload analysis will include an ASME overpressure analysis based on the cycle-specific core configuration; and
- The analysis for each fuel reload, which is current practice, confirms the capability of the system to meet the ASME design criteria.

Therefore, the requirements for nuclear system pressure relief / overpressure protection systems remain unchanged for TPO uprate conditions. The first TPO uprate cycle reload analysis will include an ASME overpressure analysis based on the cycle-specific core configuration. All safety and operational aspects of the systems are within previous evaluations.

3.2 REACTOR VESSEL

The RPV structure and support components form a pressure boundary to contain reactor coolant and moderator, and form a boundary against leakage of radioactive materials into the drywell. The RPV also provides structural support for the reactor core and internals.

3.2.1 Fracture Toughness

The TLTR, Section 5.5.1.5, describes the RPV fracture toughness evaluation process. RPV embrittlement is caused by neutron exposure of the wall adjacent to the core including the regions above and below the core that experience fluence ≥ 1.0E+17 n/cm^2. This region is defined as the "beltline" region. Operation at TPO conditions results in a higher neutron flux, which increases the integrated fluence over the period of plant life. PBAPS was evaluated for a fluence that bounds the required value for operation at TPO conditions.

The neutron fluence for TPO was calculated by [[ ]], which used two-dimensional neutron transport theory. The neutron transport methodology is consistent with RG 1.190 (Reference 22). A bounding 1/4T fluence of 1.14E+18 n/cm^2 for Unit 2 and 1.09E+18 n/cm^2 for Unit 3 is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G (Reference 23). The results of these evaluations indicate that:

(a) The upper shelf energy (USE) will remain > 50 ft-lb for the design life of the vessel, or will maintain the equivalent margin required by 10 CFR 50, Appendix G as defined in RG 1.99 (Reference 24). Many of the PBAPS RPV materials do not have sufficient unirradiated USE data. Therefore, equivalent margin analyses (EMAs) were performed for the limiting beltline plate, weld, and nozzle forging materials to ensure qualification. These values are provided in Tables 3-1a and 3-1b for PBAPS.

(b) The beltline material RT_{NDT} remains below the 200°F screening criteria as defined in RG 1.99 (Reference 24). These values are provided in Tables 3-2a and 3-2b for PBAPS.

(c) The 54 effective full power years (EFPY) TPO analyses resulted in a minor increase in adjusted reference temperature (ART) of less than 0.5°F. Additionally, the 1/4T fluence levels for TPO are slightly higher than CLTP levels as shown in the M+SAR (Reference 15). The currently licensed pressure-temperature (P-T) curves are valid for up to 54 EFPY at CLTP. These same curves have been evaluated for TPO and are conservatively limited to be valid for up to 53 EFPY. This evaluation represents a re-assessment of the duration for which the previously approved P-T curves are applicable from the duration contained in the satisfied commitment for the Time Limiting Aging Analysis (TLAA) required by the NRC in the SER for Renewed Facility Operating Licenses (DPR-44 and DPR-56) (Reference 25) and as incorporated into UFSAR License Renewal Supplement Appendix Q. It is noted that the 53 EFPY limitation accommodates a 60-year license which is currently conservatively estimated to encompass less than 50 EFPY for both units. Therefore, the currently licensed P-T curves do not require revision for the TPO uprate. This conclusion includes the effects of the N16 water level instrumentation nozzle that occurs within the beltline region.

(d) The surveillance program consists of three capsules in PBAPS Unit 2. The first capsule (at the 120° azimuthal location) was removed after Fuel Cycle 7 (after 7.53 EFPY) and tested. A reconstituted capsule was installed during the Unit 2 Refueling Outage 8 (2RO8) prior to Cycle 9, which began on April 18, 1991. The second capsule (at the 30° azimuthal location) is scheduled for removal at 33.7 EFPY. The third capsule is considered to be standby. PBAPS Unit 2 is participating in the integrated surveillance program (ISP) and is a representative host plant. TPO has no effect on the existing surveillance schedule.

(e) The surveillance program consists of three capsules in PBAPS Unit 3. The first capsule (at the 30° azimuthal location) was removed after Fuel Cycle 7 (after 7.57 EFPY) and tested. A reconstituted capsule was installed during the Unit 3 Refueling Outage 8 (3RO8) prior to Cycle 9, which began on January 2, 1992. The second and third capsules are considered to be standby. PBAPS Unit 3 is participating in the ISP and is not a representative host plant. TPO has no effect on the existing surveillance schedule.

(f) The 54 EFPY beltline circumferential weld material RT_{NDT} remains bounded by the requirements of Generic Letter (GL) 98-05 (Reference 26), Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 (References 27 and 28), and BWRVIP-74-A (References 29 and 30). This comparison is provided in Tables 3-3a and 3-3b for circumferential welds.

(g) The 54 EFPY beltline axial weld material RT_{NDT} remains bounded by the requirements of BWRVIP-74-A (Reference 30) and Reference 31. This comparison is provided in Tables 3-4a and 3-4b for axial welds.

(h) An evaluation on brittle fracture of the RPV due to reflood following a postulated LOCA was performed. The analysis shows that when the peak stress intensity occurs at approximately 300 seconds after the LOCA, the temperature of the vessel wall at 1/4T is approximately 400°F. The RPV reflood thermal shock, following a postulated LOCA, is evaluated for the maximum ART value for TPO. The evaluation calculates the temperature required to achieve a fracture toughness of 200 ksi√in when using the equation for fracture toughness stress intensity for crack initiation (K_{Ic}) presented in Appendix A of ASME Section XI. The calculated temperatures (168.9°F for Unit 2 and 193.4°F for Unit 3) are well below the minimum 400°F temperature predicted for the thermal shock event at the time of peak stress intensity.

The maximum normal operating dome pressure for TPO is unchanged from that for CLTP power operation. Therefore, the hydrostatic and leakage test pressures and associated temperatures are acceptable for the TPO. Because the vessel is still in compliance with the regulatory requirements as demonstrated above, operation at TPO does not have an adverse effect (not exceeding regulatory requirements) on the reactor vessel fracture toughness.

3.2.2 Reactor Vessel Structural Evaluation

The stress reconciliation for CLTP, considering 60-year plant license, was [[

]] the actual TPO operating power level of 4,016 MWt.

The TLTR (Reference 1) provides a disposition for [[

]].

The following table provides the justification for confirming the TLTR disposition:

Topic	TLTR Parameter(s) or Requirement(s)	Justification / CLTP versus TPO Comparison
[[		]]

[[

]]

High and low pressure seal leak detection nozzles were not considered to be pressure boundary components at the time that the OLTP evaluation was performed and have not been evaluated for TPO because they are not part of the pressure boundary region.

The effect of TPO was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME B&PV Code. For the components under consideration, the 1965 Edition with addenda to and including the Winter 1965 Addenda is applicable. However, if a component's design has been modified, the governing code for that component was the code used in the stress analysis of the modified component. There are no components that [[ ]] and were modified since the original construction.

Typically, new stresses are determined by [[ ]]. The bounding analyses were performed for the design, normal and upset, and emergency and faulted conditions. If there is an [[ ]] are considered in the analysis of the components affected for normal, upset, emergency and faulted conditions.

3.2.2.1 Design Conditions

Because there are no changes in the design conditions due to TPO, the design stresses are unchanged and the ASME Code requirements are met.

3.2.2.2 Normal and Upset Conditions

The reactor coolant temperature and flow at TPO conditions are unchanged from those at current rated conditions because the 120% of OLTP power uprate evaluations were performed at conditions [[ ]] that bound the change in operating conditions from CLTP to TPO. The evaluation type is mainly reconciliation of the stresses and usage factors to reflect TPO conditions. Calculations for TPO were not required as TPO is bounded by the evaluated EPU and MELLLA+ conditions. The PBAPS analysis results for EPU and MELLLA+ show that all components meet their ASME Code requirements and no further analysis is required.

3.2.2.3 Emergency and Faulted Conditions

The stresses due to emergency and faulted conditions are based on loads such as peak dome pressure, which are unchanged for TPO. Therefore, the ASME Code requirements are met for all RPV components under emergency and faulted conditions.

As part of the TPO evaluation scope, GEH safety communications (SCs) were also considered in the reactor vessel stress evaluations. GEH SC 11-07 (Reference 32) was determined to not be applicable to PBAPS as the SC concerns are [[ ]]. GEH SC 12-20 (Reference 33) and SC 13-08 (Reference 34) were determined to be applicable to PBAPS. As a result, the shroud support to the RPV connection region stress evaluation was reconciled to consider an [[ ]] acoustic loads. As shown in Table 3-5, the shroud support (attachment to RPV) component was shown to be within the allowable limits and demonstrated to be structurally qualified for operation at TPO conditions when reconciled to incorporate GEH SC 12-20 (Reference 33) and SC 13-08 (Reference 34) concerns.

3.3 REACTOR INTERNALS

The reactor internals include core support structure (CSS) and non-core support structure (non-CSS) components.

3.3.1 Reactor Internal Pressure Differences

The reactor internal pressure differences (RIPDs) are affected by the maximum licensed CF rate, which is unchanged for the TPO uprate. The effect due to the changes in loads for both Normal and Upset conditions is reported in Section 3.3.2. The Normal and Upset evaluations of RIPDs for the TPO uprate are slightly increased from CLTP, which includes EPU and MELLLA+. The Emergency and Faulted evaluations of RIPDs for the TPO uprate are conservatively bounded by the EPU analysis performed at 102% of CLTP. For the steam dryer, station specific analyses for the TPO (also called measurement uncertainty recapture (MUR)) and MELLLA+ conditions were performed and found acceptable. The results of this analysis are provided in MUR LAR Attachment 10.

Fuel bundle lift margins are calculated at the Faulted condition to demonstrate that fuel bundles would not lift under the worst conditions. The current analyses conservatively assumed 102% of CLTP and 110% of CF, which bounds TPO conditions at 101.66% of CLTP and 110% of CF. The fuel lift margins for the Normal and Upset conditions at the TPO RTP are slightly decreased from

CLTP. The fuel lift margins for the Normal and Upset conditions at the TPO RTP are bounded by Emergency and Faulted conditions. The effect due to the changes in fuel lift margins is reported in Section 3.3.2.

Acoustic and flow-induced loads on the jet pump, core shroud and shroud support due to a recirculation line break (RLB) are conservatively bounded by the EPU analysis performed at 102% of CLTP.

3.3.2 Reactor Internals Structural Evaluation

The RPV internals consist of the CSS components and non-CSS components. The RPV internals are not ASME code components; however, the requirements of the ASME code are used as guidelines in their design/analysis. The evaluations/stress reconciliation in support of the TPO was performed consistent with the design basis analysis of the components. The reactor internal components evaluated are:

CSS Components

Shroud

Shroud Support

Core Plate

Top Guide

Control Rod Drive Housing

Control Rod Guide Tube (CRGT)

Orificed Fuel Support (OFS)

Fuel Channel

Non-CSS Components

Feedwater Sparger

Jet Pump Assembly

Core Spray Line and Sparger

Access Hole Cover

Steam Dryer (Refer to MUR LAR Attachment 10)

Shroud Head and Steam Separator Assembly

In-Core Housing and Guide Tube

Core Differential Pressure and Liquid Control Line

Jet Pump Instrument Penetration Seal

The original configurations of the RPV internals are considered in the TPO evaluation unless a component has undergone permanent structural modifications, in which case, the modified configuration is used as the basis for the evaluation (e.g., jet pumps).

The loads considered in the evaluation of the RPV internals include RIPDs, dead weight, seismic loads, acoustic and flow induced loads due to RLB, hydraulic flow and thermal loads.

RPV design pressure remains unchanged. RIPD loads are bounded by the 102% of CLTP values except for the shroud. The increase in the shroud RIPD is very small and deemed insignificant. The seismic load and acoustic and flow induced loads due to RLB remain unchanged for TPO. The feedwater flow and thermal load remain bounded by the 102% of CLTP values. The change in hydraulic flow is considered negligible. The effect of weight change on load due to jet pump repair is insignificant. All applicable loads remain unchanged or unaffected for the TPO condition.

All the RPV internals were shown to be within the allowable limits. The limiting stresses and fatigue usage factors of all RPV internal components are summarized in Table 3-6 and Table 3-7, respectively. Therefore, the RPV internal components are demonstrated to be structurally qualified for operation at TPO conditions.

3.3.3 Steam Separator and Dryer Performance

The performance of the PBAPS steam separator/dryer has been evaluated to determine the moisture content of the steam leaving the reactor pressure vessel. The results of the evaluation demonstrated that the steam separator/dryer performance remains acceptable (i.e., moisture content ≤ 0.10 wt. %) at TPO conditions. TPO results in an increase in the amount of saturated steam generated in the reactor core. For constant CF, this increase in the amount of saturated steam results in an increase in the average separator inlet quality and an increase in the steam dryer face velocity. These factors, in addition to the radial power distribution, affect the steam separator/dryer performance. The PBAPS plant-specific evaluation concluded that the performance of the steam dryer and separator remains acceptable in the TPO region. Carryunder performance under TPO conditions also remains acceptable.

3.4 FLOW-INDUCED VIBRATION

The process for the reactor vessel internals vibration assessment is described in TLTR Section 5.5.1.3. An evaluation determined the effects of FIV on the reactor internals at 110% of rated CF and 102% of CLTP. The vibration levels for the TPO conditions were estimated from measured vibration data during startup tests on the NRC designated prototype plant, Browns Ferry Unit 1. For components requiring an evaluation but not instrumented in Browns Ferry Unit 1, vibration data from similar plants or test facilities, or analytical results, are used. The expected vibration levels for TPO were estimated by extrapolating the measured vibration data, based on GEH BWR operating experience. These expected vibration levels were compared with established vibration acceptance limits.

The following components were evaluated for the TPO uprate:

Component(s)	Process Parameter(s)	TPO Evaluation
Core Spray Piping and Sparger	Structural natural frequencies versus vortex shedding frequency (VSF).	No VSF resonance in TPO region. No change in stress.
Feedwater Sparger	Feedwater flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV stress. Extrapolation of measured data shows that stresses are within limits.
Fuel Channels	The operating conditions used in the fuel assembly design (GNF2) are bounding for PBAPS at TPO conditions.	The PBAPS fuel assembly (GNF2) is acceptable under FIV for TPO conditions.
Guide Rods	The lock-in condition is conservatively assumed and calculated by the equation from Reference 35.	The maximum stress is within the GEH acceptance criteria of 10,000 psi.
Jet Pumps	The increase in jet pump flow at TPO is negligible based on no change in CF and a minor increase in core differential pressure (< 0.1 psi).	Slight increase (< 2%) in FIV stress. Extrapolation of measured data shows that stresses are within limits.
Jet Pump Sensing Lines (JPSLs)	Vane passing frequency of recirculation pumps.	No resonance at vane passing frequency range due to TPO. JPSL with mitigation clamps and/or a pump with limited speed prevents resonance with vane passing frequency.
Shroud	Flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV stress. Extrapolation of measured data shows that stresses are within limits.

Component(s)	Process Parameter(s)	TPO Evaluation
Shroud Head and Separator	Steam flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV stress. Extrapolation of measured data shows that stresses are within limits.
CRGT and In-Core Guide Tubes	Core flow at TPO is unchanged from CLTP.	No change in stress.
RPV Top Head Nozzles	Negligible steam flow in area.	No change in stress.

The calculations for the TPO uprate conditions indicate that vibrations of all safety-related reactor internal components are within the GEH acceptance criteria. The analysis is conservative for the following reasons:

The GEH criteria of 10,000 psi peak stress intensity is more conservative than the ASME allowable peak stress intensity of 13,600 psi for service cycles $\geq 10^{11}$.

Conservatively, the peak responses of the applicable modes are absolute summed.

The maximum vibration stress amplitude of each mode is used in the absolute sum process, whereas in reality the maximum vibration amplitudes are unlikely to occur at the same time.

The flow-induced vibration evaluation for the replacement steam dryer is provided in MUR LAR Attachment 10 and concludes that the stresses at TPO and MELLLA+ conditions remain within acceptance limits.

Therefore, it is concluded that the flow-induced vibrations for all evaluated components remain within acceptance limits.

The piping components were evaluated in accordance with ASME Code N-1300 (Reference 35) FIV analysis guidelines. The resonance separation rule in ASME Appendix N Subparagraph N-1324.1(d) was used to determine if adequate separation exists between the vortex shedding frequencies and the natural frequencies of the piping components.

The safety-related main steam (MS) and FW piping flow rates increase less than 2% due to the TPO uprate. The reactor recirculation system (RRS) flow rate is essentially unchanged at TPO. The MS and FW piping thermowells experience increased vibration levels, approximately proportional to the increase in the square of the flow velocities and in proportion to any increase in fluid density. The decrease in fluid density for TPO conditions, as the result of about a 2°F increase in FW temperature, is insignificant. Analytical evaluations have shown that the safety-related piping components, and thermowells, in the MS, FW, and RRS piping are structurally adequate for TPO conditions.

The MS and FW piping experiences increased vibration levels approximately proportional to the increase in the square of the flow velocities and in proportion to any increase in fluid density.

The MS piping vibration is expected to increase only by about 4% due to the increase in its flow rate from 4.043 Mlb/hr per line at CLTP to 4.121 Mlb/hr per line at the TPO bounding thermal power of 4,018 MWt. A MS piping FIV test program, after the implementation of the power uprate to CLTP, showed that vibration levels were within acceptance criteria, and operating experience shows that there are no existing vibration problems in MS lines at CLTP operating conditions. An assessment of the approximately 4% increase in vibration level concludes that MS piping vibrations will remain within acceptance limits at TPO conditions.

Similarly, the FW piping vibration is expected to increase only by about 4% due to the increase in its total flow rate from 16.139 Mlb/hr at CLTP to 16.453 Mlb/hr at the TPO bounding thermal power of 4,018 MWt. A FW piping FIV test program, after the implementation of the power uprate to CLTP, showed that vibration levels were within acceptance criteria, and operating experience shows that there are no existing vibration problems in FW lines at CLTP operating conditions. An assessment of the approximately 4% increase in vibration level concludes that FW piping vibrations will remain within acceptance limits at TPO conditions.

The change in fluid density for TPO conditions, as the result of an approximately 2ºF increase in FW temperature, is insignificant.

The RRS flow rate, thus velocity, is unchanged from CLTP to TPO RTP; therefore, the RRS piping vibration is unaffected.

3.5 PIPING EVALUATION

3.5.1 Reactor Coolant Pressure Boundary Piping

The methods used for the plant-specific piping and pipe support evaluations are identical to those used in the PBAPS EPU (Reference 14). The effect of the TPO uprate with no nominal vessel dome pressure increase is negligible for the reactor coolant pressure boundary (RCPB) portion of all piping except for portions of the FW lines, MS lines, and piping connected to the FW and MS lines. The following table summarizes the evaluation of the piping inside containment.

Component(s) / Concern	Process Parameter(s)	TPO Evaluation
Recirculation System Pipe Stresses Pipe Supports	Nominal dome pressure at TPO RTP is identical to CLTP. Recirculation flow at TPO RTP is identical to CLTP. Small increase in core pressure drop of < 1 psi. Recirculation fluid temperature changes by < 1ºF.	Current licensing basis envelops TPO conditions; therefore, the piping system is acceptable for TPO. Negligible change in pipe stress. Negligible effect on pipe supports.

Component(s) / Concern	Process Parameter(s)	TPO Evaluation
MS and Attached Piping (Inside Containment) (e.g., SRV discharge line piping up to first anchor, reactor core isolation cooling (RCIC) / high pressure coolant injection (HPCI) piping, MS drain lines, RPV head vent line piping located inside containment) Pipe Stresses Pipe Supports	Nominal dome pressure at TPO RTP is identical to CLTP. Steam flow at TPO RTP is ~2% greater than CLTP. No change in main steam line (MSL) pressure.	Current licensing basis envelops TPO conditions; therefore, the piping system is acceptable for TPO. Minor change in pipe stress. Minor effect on pipe supports.
FW and Attached Piping (Inside Containment) Pipe Stresses Pipe Supports	Nominal dome pressure at TPO RTP is identical to CLTP. FW flow at TPO RTP is ~2% greater than CLTP. Minor change in FW line pressure (< 2% increase). Minor change in FW temperature (~ 2°F increase).	Current licensing basis envelops TPO conditions; therefore, the piping system is acceptable for TPO. Negligible change in pipe stress. Negligible effect on pipe supports.
RPV Bottom Head Drain Line, RCIC Piping, HPCI Piping, Low Pressure Coolant Injection (LPCI) Piping, Core Spray (CS) Piping, SLCS Piping, and Reactor Water Cleanup (RWCU) Piping Pipe Stresses Pipe Supports	Nominal dome pressure at TPO RTP is identical to CLTP. Small change in pressure of < 1%. Recirculation fluid temperature changes < 1°F.	Current licensing basis envelops TPO conditions; therefore, the piping system is acceptable for TPO. Negligible change in pipe stress. Negligible effect on pipe supports.

For the MS and FW lines, supports, and connected lines, the methodologies as described in the PBAPS PUSAR (Reference 14) were used to determine the percent increases in applicable ASME code stresses, displacements, cumulative usage factors (CUFs), and pipe interface component loads (including supports) as a function of percentage increase in pressure (where applicable), temperature, and flow due to TPO conditions. The percentage increases were applied to the highest calculated stresses, displacements, and the CUF at applicable piping system node points to conservatively determine the maximum TPO calculated stresses, displacements and usage factors. This approach is conservative because the TPO does not affect weight and all building filtered loads (i.e., seismic loads are not affected by the TPO). The factors were also applied to nozzle loads, support loads, penetration loads, valves, pumps, heat exchangers and anchors so that these components could be evaluated for acceptability, where required. No new computer codes were used or new assumptions introduced for this evaluation.

MS and Attached Piping System Evaluation

The MS piping system (inside containment) was evaluated for compliance with the ASME code stress criteria, and for the effects of thermal displacements on the piping snubbers, hangers, and struts. Piping interfaces with RPV nozzles, penetrations, flanges and valves were also evaluated.

Pipe Stresses

The evaluation shows that the increase in flow associated with the TPO uprate does not result in load limits being exceeded for the MS piping system or for the RPV nozzles. The original design analyses have sufficient design margin between calculated stresses and ASME code allowable limits to justify operation at the TPO uprate conditions. The temperature of the MS piping (inside containment) is unchanged for the TPO.

The design adequacy evaluation results show that the requirements of American National Standards Institute (ANSI) USAS B31.1, B31.7 Power Piping and ASME, Section III, Subsection ND (as applicable) requirements are satisfied for the evaluated piping systems. Therefore, the TPO does not have an adverse effect on the MS piping design.

Pipe Supports

The MS piping was evaluated for the effects of transient loading on the piping snubbers, hangers, struts, and pipe whip restraints. A review of the increases in MS flow associated with the TPO uprate indicates that piping load changes do not result in any load limit being exceeded.

Erosion / Corrosion

The carbon steel MS piping can be affected by flow-accelerated corrosion (FAC). FAC is affected by changes in fluid velocity, temperature and moisture content. PBAPS has an established FAC monitoring program, also evaluated by the NRC for license renewal and EPU, for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. The variation in velocity, temperature, and moisture content resulting from the TPO uprate are minor changes to parameters affecting FAC. The FAC monitoring program includes the use of a predictive modeling program, EPRI CHECWORKSTM SFA 3.0, to calculate wall thinning of components susceptible to FAC. For TPO, the evaluation of predicted wall thinning of the MS and attached piping indicates negligible differences in comparison to the EPU evaluation.

No changes to piping inspection scope and frequency are required prior to TPO implementation to ensure adequate margin for the changing process conditions. The FAC monitoring program provides assurance that effects from TPO on high energy piping systems potentially susceptible to pipe wall thinning due to FAC are monitored and addressed.

FW Piping System Evaluation

The FW piping system (inside containment) was evaluated for compliance with the ASME code stress criteria, and for the effects of thermal expansion displacements on the piping snubbers, hangers, and struts. Piping interfaces with RPV nozzles, penetrations, and valves were also evaluated.

Pipe Stresses

A review of the small increases in temperature, pressure, and flow associated with the TPO uprate indicates that piping load changes do not result in load limits being exceeded for the FW piping system or for RPV nozzles. The original design analyses have sufficient design margin between calculated stresses and ASME code allowable limits to justify operation at the TPO uprate conditions.

The design adequacy evaluation shows that the requirements of ANSI (USAS) B31.1, B31.7 Power Piping and ASME, Section III, Subsection ND-3600 requirements remain satisfied. Therefore, the TPO does not have an adverse effect on the FW piping design.

Pipe Supports

The TPO does not affect the FW piping snubbers, hangers, struts, and pipe whip restraints. A review of the increase in FW temperature and flow associated with the TPO indicates that piping load changes do not result in any load limit being exceeded at the TPO uprate conditions.

Erosion / Corrosion

The carbon steel FW piping is susceptible to FAC. TPO affects FAC in the FW piping via changes in fluid velocity and temperature. PBAPS has an established program for monitoring pipe wall thinning in single and two-phase high energy carbon steel piping. The FAC monitoring program, also evaluated by the NRC for license renewal and EPU, includes the use of a predictive modeling program, EPRI CHECWORKS™ SFA 3.0, to calculate wall thinning of components susceptible to FAC. The variation in velocity and temperature resulting from the TPO uprate are minor changes to parameters affecting FAC. At TPO conditions, the evaluation of predicted wall thinning of the FW piping system indicates negligible differences in comparison to the EPU evaluation.

No changes to piping inspection scope and frequency are required prior to TPO implementation to ensure adequate margin exists for the TPO process conditions. The FAC monitoring program provides assurance that any adverse effect from TPO on high energy piping systems potentially susceptible to pipe wall thinning due to FAC is monitored and addressed.

3.5.2 Balance-of-Plant Piping Evaluation

This section addresses the adequacy of the BOP piping design (outside of the RCPB) for operation at the TPO conditions.

Pipe Supports

For the condensate, FW, extraction steam (ES), heater drain, and MS systems, operating system pressures and temperatures under TPO will remain within design ratings.

Because there is no change in the MS operating temperature from the reactor to the MS stop valves, there is no change in the thermal expansion stress for TPO. For systems with increased operating temperatures (i.e., MS downstream of the stop valves, condensate, FW, ES, heater drains), changes to thermal expansion stresses are small and acceptable. Pipe support loads will experience a small increase in thermal loads (<1%). However, when considering the combination with other loads that are not affected by the TPO uprate (e.g., deadweight) the combined support load increase is insignificant. This piping has been analyzed to conditions which envelope operations under TPO.

For the MS system piping outside containment, the existing TSV closure transient analysis was reviewed and determined to bound the TPO uprate conditions. No new piping analysis was required.

Erosion / Corrosion

The integrity of high energy piping systems is assured by proper design in accordance with the applicable codes and standards. Piping thickness of carbon steel components can be affected by FAC. PBAPS has an established program for monitoring pipe wall thinning in single phase and two-phase high energy carbon steel piping. The variation in velocity and temperature resulting from the TPO uprate are minor changes to parameters affecting FAC. The FAC monitoring program, also evaluated by the NRC for license renewal and EPU, includes the use of a predictive modeling program, EPRI CHECWORKS™ SFA 3.0, to calculate wall thinning of components susceptible to FAC. For TPO, the evaluation of predicted wall thinning of the BOP piping indicates a negligible difference in comparison to the EPU evaluation.

Operation at the TPO RTP results in some changes to parameters affecting FAC in those systems associated with the turbine cycle (e.g., condensate, FW, MS). The evaluation of the need for inspection for FAC in BOP systems is addressed by compliance with GL 89-08 (Reference 36). Continued monitoring of the affected systems through the FAC program provides confidence in the integrity of susceptible high energy piping systems. No changes to piping inspection frequency are required to ensure adequate margin exists for those systems with changing process conditions as a result of TPO. Continued implementation of existing procedures to manage the PBAPS FAC program provides assurance that any adverse effect from TPO on high energy piping systems potentially susceptible to pipe wall thinning due to FAC is monitored and addressed.

3.6 REACTOR RECIRCULATION SYSTEM

A plant-specific evaluation was performed for the PBAPS RRS using the evaluation approach presented in TLTR Section 5.6.2. The TPO uprate has a minor effect on the RRS and its components. Operation at the TPO uprated power is accomplished along an extension of the current MELLLA+ boundary with no increase in the maximum CF. No significant reduction of the maximum flow capability occurs due to the TPO uprate because of the small increase in core pressure drop of 0.01 psid. The effect on pump net positive suction head (NPSH) at TPO conditions is negligible. An evaluation has confirmed that no significant increase in RRS vibration occurs from the TPO operating conditions.

The cavitation protection interlock for the recirculation pumps and jet pumps is expressed in terms of FW flow. This interlock is based on sub-cooling and thus is a function of absolute FW flow rate and FW temperature at less than full thermal power operating conditions. Therefore, the interlock is not changed by TPO.

3.7 MAIN STEAM LINE FLOW RESTRICTORS

The plant-specific EPU evaluation (Reference 14), using the approach described in TLTR Appendix J.2.3.7, was previously performed at 102% of CLTP.

A plant-specific evaluation of the effects of TPO RTP compared to CLTP determined that there is:

- No change in operating temperature;
- No change in maximum operating dome pressure, therefore, the resulting break flow rate is unchanged;
- A slight decrease in operating pressure along the steam line due to the higher flow rate pressure drop; and
- Less than 2% change in normal steam flow.

The plant-specific evaluation also concludes that:

- There is no increase in the steam flow for a main steam line break accident (MSLBA) because the flow restrictor and operating pressure remains unchanged;
- Because the flow restrictors were designed and analyzed for the choke flow condition with the maximum pressure difference, which is bounding for the TPO uprate condition, the structural integrity of the MSL flow restrictors are not affected by a TPO uprate; and
- The less than 2% change in normal steam flow does not affect any accident-related loads because the current loads continue to bound the analysis for TPO uprate operation.

Therefore, the requirements for the MSL flow restrictors remain unchanged for TPO uprate conditions. All safety and operational aspects of the MSL flow restrictors are within previous evaluations.

3.8 MAIN STEAM ISOLATION VALVES

The plant-specific EPU evaluation (Reference 14), using the approach described in TLTR Appendix J.2.3.7, was previously performed at 102% of CLTP.

A plant-specific evaluation of the effects of TPO RTP compared to CLTP determined that there is:

- No change in operating temperature;
- A slight decrease in operating pressure along the steam line due to the higher flow rate pressure drop; and
- Less than 2% change in normal steam flow.

The plant-specific evaluation also concludes that:

- There is no increase in the steam flow for an MSLBA because the flow restrictor and operating pressure remains unchanged; and
- The less than 2% change in normal steam flow does not affect any accident-related loads because the current loads continue to bound the analysis for TPO uprate operation.

Therefore, the requirements for the MSIVs remain unchanged for TPO uprate conditions. All safety and operational aspects of the MSIVs are within previous evaluations.

3.9 REACTOR CORE ISOLATION COOLING

The RCIC system provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high pressure makeup systems.

The plant-specific evaluation specifically determined that there is no change in the following:

- Operating pressure;
- Pressure setpoints of the SRVs;
- The capability of the turbine-driven RCIC system to successfully develop the horsepower and speed required by the pumps; and
- RCIC capacity.

The plant-specific evaluation also concludes that:

- The LOFW AOR, including decay heat inputs, which was performed at 102% of CLTP, bounds the TPO uprate operating conditions; and
- The capability to maintain the water level above the top of active fuel (TAF) remains unchanged.

The conclusion in the LOFW analysis-of-record based on SAFER/GSTRM will remain valid with SAFER/PRIME as the water level response between the SAFER/GSTRM and the SAFER/PRIME methodologies are expected to be essentially the same. The minimum level is maintained at least 129 inches above TAF in the analysis-of-record.

The TPO uprate does not affect the RCIC system operation, initiation, or capability requirements.

3.10 RESIDUAL HEAT REMOVAL SYSTEM

The residual heat removal (RHR) system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to function in several operating modes. Plant-specific evaluations were performed for PBAPS using the evaluation approaches provided in TLTR Section 5.6.4 and Appendices J.2.3.1 and J.2.3.13.

The following table summarizes the effect of the TPO on the design basis of the RHR system.

Operating Mode	Key Function	TPO Evaluation
LPCI Mode	Reactor vessel coolant makeup.	See Section 4.2.3.
Containment Cooling	Normal suppression pool cooling (SPC) function is to maintain pool temperature below the limit. For abnormal events or accidents, the SPC mode maintains the long-term pool temperature below the DL. The containment spray cooling (CSC) mode sprays water into the containment to reduce post-accident containment pressure and temperature. The coolant injection cooling mode establishes alternate shutdown cooling by injecting water into the reactor vessel, which is cooled with two RHR heat exchangers, and reduces containment temperature.	Containment analyses have been performed at 102% of CLTP, which bounds TPO.
Shutdown Cooling (SDC) Mode	Removes sensible and decay heat from the reactor primary system during a normal reactor shutdown.	The slightly higher decay heat has a negligible effect on the SDC mode, which has no safety function.
Steam Condensing Mode	Decay heat removal.	PBAPS does not have a steam condensing mode of RHR.
Fuel Pool Cooling (FPC) Assist	Supplements FPC in the event that the fuel pool heat load exceeds the heat removal capability of the FPC system.	See Section 6.3.1.

The ability of the RHR system to perform required safety functions is demonstrated with analyses based on 102% of CLTP. Therefore, the plant-specific evaluation concludes that all safety aspects of the RHR system are within previous evaluations. The requirements for the RHR system remain unchanged for TPO uprate conditions.

3.11 REACTOR WATER CLEANUP SYSTEM

A plant-specific evaluation of the RWCU system was performed for PBAPS using the evaluation approach provided in TLTR Section 5.6.6 and Appendix J.2.3.4.

The plant-specific evaluation specifically verified that:

- There is no significant change in nominal operating temperature (< 1°F) in the high pressure portion of the system for the PBAPS TPO uprate;
- There is no change in nominal operating pressure in the high pressure portion of the system for the PBAPS TPO uprate;
- There is no identifiable change in the level of impurities in the reactor water with respect to any effect upon regeneration frequency;
- The capacity of the RWCU system is sufficient, possibly with small operational adjustments, to accommodate the small effect that the TPO uprate is expected to have on RWCU duty; and
- The FW system iron input is not increased significantly by TPO operation.

The plant-specific evaluation concludes that the performance requirements of the RWCU system are negligibly affected by TPO uprate; therefore, the safety and operational aspects of water chemistry performance are not affected by the TPO.

Table 3-1a Upper Shelf Energy EMA for TPO 60-Year License (54 EFPY) – Unit 2

Equivalent Margin Analysis Plant Applicability Verification Form PBAPS Unit 2 Including TPO Conditions 60-Year License (54 EFPY)			
BWR/3-6 Plate			
ISP Surveillance Plate USE (Heat C2761-2):			
[[		]]	
Unirradiated USE	=	127.2 ft-lb	
1st Capsule Measured USE	=	133.0 ft-lb	
1st Capsule Fluence	=	1.8E+17 n/cm^2	
1st Capsule Measured % Decrease	=	-4.6	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	8.0	(RG 1.99, Revision 2, Figure 2)
Limiting Beltline Plate USE (Heat C2894-2):			
%Cu	=	0.13	
54 EFPY 1/4T Fluence	=	1.14E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	13.5	(RG 1.99, Revision 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Revision 2, Position 2.2)
13.5%	≤	[[ ]]	
Therefore, vessel plates are bounded by the EMA.			

Equivalent Margin Analysis Plant Applicability Verification Form PBAPS Unit 2 Including TPO Conditions 60-Year License (54 EFPY)			
BWR/2-6 Weld			
ISP Surveillance Weld USE (Heat PB2 ESW):			
[[%Cu		]]	
Unirradiated USE	=	110.9 ft-lb	
1st Capsule Measured USE	=	113.6 ft-lb	
1st Capsule Fluence	=	1.8E+17 n/cm^2	
1st Capsule Measured % Decrease	=	-2.4	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	9.5	(RG 1.99, Revision 2, Figure 2)
Limiting Beltline Weld USE (Heat 37C065):			
%Cu	=	0.182	
54 EFPY 1/4T Fluence	=	1.14E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	20.0	(RG 1.99, Revision 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Revision 2, Position 2.2)
20.0%	≤	[[ ]]	
Therefore, vessel welds are bounded by the EMA.			

Table 3-1b Upper Shelf Energy EMA for TPO 60-Year License (54 EFPY) – Unit 3

Equivalent Margin Analysis Plant Applicability Verification Form PBAPS Unit 3 Including TPO Conditions 60-Year License (54 EFPY)			
BWR/3-6 Plate			
ISP Surveillance Plate USE (Heat B0673-1):			
%Cu	=	0.15	
Unirradiated USE	=	158.1 ft-lb	
1st Capsule Measured USE	=	158.8 ft-lb	
1st Capsule Fluence	=	5.09E+17 n/cm^2	
2nd Capsule Measured USE	=	137.0 ft-lb	
2nd Capsule Fluence	=	1.17E+18 n/cm^2	
3rd Capsule Measured USE	=	133.0 ft-lb	
3rd Capsule Fluence	=	1.87E+18 n/cm^2	
4th Capsule Measured USE	=	131.3 ft-lb	
4th Capsule Fluence	=	2.63E+18 n/cm^2	
1st Capsule Measured % Decrease	=	-0.4	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	12.0	(RG 1.99, Revision 2, Figure 2)
2nd Capsule Measured % Decrease	=	13.4	(Charpy Curves)
2nd Capsule RG 1.99 Predicted % Decrease	=	14.5	(RG 1.99, Revision 2, Figure 2)
3rd Capsule Measured % Decrease	=	15.9	(Charpy Curves)
3rd Capsule RG 1.99 Predicted % Decrease	=	16.5	(RG 1.99, Revision 2, Figure 2)
4th Capsule Measured % Decrease	=	17.0	(Charpy Curves)
4th Capsule RG 1.99 Predicted % Decrease	=	18.0	(RG 1.99, Revision 2, Figure 2)
Limiting Beltline Plate USE (Heat C2773-2):			
%Cu	=	0.15	
54 EFPY 1/4T Fluence	=	1.09E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	14.5	(RG 1.99, Revision 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Revision 2, Position 2.2)
14.5%	≤	[[ ]]	
Therefore, vessel plates are bounded by the EMA.			

Table 3-1b Upper Shelf Energy EMA for TPO 60-Year License (54 EFPY) – Unit 3 (continued)

Equivalent Margin Analysis Plant Applicability Verification Form PBAPS Unit 3 Including TPO Conditions 60-Year License (54 EFPY)			
BWR/2-6 Weld			
ISP Surveillance Weld USE (Heat 5P6756):			
%Cu	=	0.06	
Unirradiated USE	=	104.4 ft-lb	
1st Capsule Measured USE	=	84.4 ft-lb	
1st Capsule Fluence	=	1.16E+18 n/cm^2	
2nd Capsule Measured USE	=	79.3 ft-lb	
2nd Capsule Fluence	=	1.94E+18 n/cm^2	
3rd Capsule Measured USE	=	84.6 ft-lb	
3rd Capsule Fluence	=	1.58E+18 n/cm^2	
4th Capsule Measured USE	=	110.7 ft-lb	
4th Capsule Fluence	=	2.93E+17 n/cm^2	
1st Capsule Measured % Decrease	=	19.2	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	12.1	(RG 1.99, Revision 2, Figure 2)
2nd Capsule Measured % Decrease	=	24.0	(Charpy Curves)
2nd Capsule RG 1.99 Predicted % Decrease	=	13.7	(RG 1.99, Revision 2, Figure 2)
3rd Capsule Measured % Decrease	=	19.0	(Charpy Curves)
3rd Capsule RG 1.99 Predicted % Decrease	=	13.0	(RG 1.99, Revision 2, Figure 2)
4th Capsule Measured % Decrease	=	-6.0	(Charpy Curves)
4th Capsule RG 1.99 Predicted % Decrease	=	8.8	(RG 1.99, Revision 2, Figure 2)
Limiting Beltline Weld USE (Heat 37C065):			
%Cu	=	0.182	
54 EFPY 1/4T Fluence	=	1.09E+18 n/cm^2	
RG 1.99 Predicted % Decrease	=	19.5	(RG 1.99, Revision 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Revision 2, Position 2.2)
19.5%	≤	[[ ]]	
Therefore, vessel welds are bounded by the EMA.			

Table 3-2a Adjusted Reference Temperatures for TPO 60-Year License (54 EFPY) – Unit 2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches = 6.125

54 EFPY Peak I.D. Fluence = 1.65E+18 n/cm^2
54 EFPY Peak 1/4T Fluence = 1.14E+18 n/cm^2

Lower Shell Plates, Circumferential Weld and Axial Welds

Thickness in inches = 6.125

54 EFPY Peak I.D. Fluence = 1.24E+18 n/cm^2
54 EFPY Peak 1/4T Fluence = 8.59E+17 n/cm^2

Water Level Instrumentation Nozzle (Lower-Intermediate Shell)

Thickness in inches = 6.125

54 EFPY Peak I.D. Fluence = 4.81E+17 n/cm^2
54 EFPY Peak 1/4T Fluence = 3.33E+17 n/cm^2

Component	Heat	% Cu	% Ni	ChF[1]	Adjusted ChF	Initial RT_{NDT} °F	1/4T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES Plates:													
Lower Shell Mark 57	C2791-2	0.12	0.52	81.4		-8	8.59E+17	31.5	0	15.8	31.5	63.0	55.0
	C2761-1	0.11	0.54	73.4		-14	8.59E+17	28.4	0	14.2	28.4	56.8	42.8
	C2873-2	0.12	0.57	82.4		-20	8.59E+17	31.9	0	15.9	31.9	63.8	43.8
Lower-Intermediate Shell Mark 58	C2894-2	0.13	0.42	85.6		-20	1.14E+18	38.0	0	17.0	34.0	72.0	52.0
	C2873-1	0.12	0.57	82.4		-6	1.14E+18	36.6	0	17.0	34.0	70.6	64.6
	C2761-2	0.11	0.54	73.4		-20	1.14E+18	32.6	0	16.3	32.6	65.2	45.2
Axial Welds:													
Lower Shell B1, B2, B3	37C065	0.182	0.181	94.5		-45	8.59E+17	36.6	16	18.3	48.6	85.2	40.2
Lower-Int Shell C1, C2, C3	37C065	0.182	0.181	94.5		-45	1.14E+18	42.0	16	21.0	52.8	94.7	49.7

Table 3-2a Adjusted Reference Temperatures for TPO 60-Year License (54 EFPY) – Unit 2 (continued)

Component	Heat	% Cu	% Ni	ChF[1]	Adjusted ChF	Initial RT_{NDT} °F	1/4T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
Circumferential Welds:													
BC	S-3986 Linde 124 Lot 3876	0.056	0.96	76.4		-32	8.59E+17	29.6	0	14.8	29.6	59.1	27.1
Nozzles:													
N16 [2]	C2873-1	0.12	0.57	82.4		-6	3.33E+17	19.2	0	9.6	19.2	38.5	32.5
Best Estimate Chemistries from BWRVIP-135 R3 (Reference 37):													
BC	S-3986	[[	]]	79.2		-32	8.59E+17	30.7	0	15.3	30.7	61.3	29.3
Integrated Surveillance Program (BWRVIP-135 R3):													
Plate [3] Weld [4]	C2761-2 PB2 ESW	[[	]]	65.0 84.2		-20 -45	1.14E+18 1.14E+18	28.9 37.4	0 0	14.4 18.7	28.9 37.4	57.7 74.8	37.7 29.8

Notes:

1. ChF = Chemistry Factor.
2. The N16 water level instrumentation nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur. The weld connecting the forging to the vessel shell is also Alloy 600 material, and is not required to be evaluated.
3. The ISP plate material is not the vessel target material, but does occur within the Unit 2 beltline region (Lower-Intermediate Shell). Therefore, this material is considered in determining the limiting ART. Only one set of surveillance data is currently available; therefore, upon testing of a second ISP capsule scheduled for 2018, the ChF can be reviewed.
4. The ISP weld material is not the vessel target material and does not occur within the Unit 2 beltline region. Therefore, this material is not considered in determining the limiting ART. The ChF is determined using RG 1.99 for the ISP chemistry.

Table 3-2b Adjusted Reference Temperatures for TPO 60-Year License (54 EFPY) – Unit 3

Intermediate Shell Plates, Axial Welds, and Circumferential Weld (EF)

Thickness in inches = 6.125

54 EFPY Peak I.D. fluence = 9.65E+17 n/cm^2
54 EFPY Peak 1/4T fluence = 6.68E+17 n/cm^2

Lower-Intermediate Shell Plates and Axial Welds

Thickness in inches = 6.125

54 EFPY Peak I.D. fluence = 1.57E+18 n/cm^2
54 EFPY Peak 1/4T fluence = 1.09E+18 n/cm^2

Lower Shell Plates, Circumferential Welds, Axial Welds, and Circumferential Weld (DE)

Thickness in inches = 6.125

54 EFPY Peak I.D. fluence = 9.32E+17 n/cm^2
54 EFPY Peak 1/4T fluence = 6.45E+17 n/cm^2

Water Level Instrumentation Nozzle (Intermediate Shell)

Thickness in inches = 6.125

54 EFPY Peak I.D. fluence = 4.57E+17 n/cm^2
54 EFPY Peak 1/4T fluence = 3.16E+17 n/cm^2

Component	Heat	% Cu	% Ni	ChF	Adjusted ChF	Initial RT_{NDT} °F	1/4T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
PLANT-SPECIFIC CHEMISTRIES													
Plates:													
Lower Shell													
6-146-1	C4689-2	0.12	0.56	82.2		-10	6.45E+17	27.5	0	13.8	27.5	55.1	45.1
6-146-3	C4684-2	0.13	0.58	90.4		-20	6.45E+17	30.3	0	15.1	30.3	60.6	40.6
6-146-7	C4627-1	0.12	0.57	82.4		-20	6.45E+17	27.6	0	13.8	27.6	55.2	35.2
Lower-Intermediate Shell													
6-139-10	C2773-2	0.15	0.49	104.0		10	1.09E+18	45.1	0	17.0	34.0	79.1	89.1
6-139-11	C2775-1	0.13	0.46	86.8		10	1.09E+18	37.7	0	17.0	34.0	71.7	81.7
6-139-12	C3103-1	0.14	0.6	100.0		10	1.09E+18	43.4	0	17.0	34.0	77.4	87.4
Intermediate Shell													
6-146-5	C4608-1	0.12	0.55	82.0		10	6.68E+17	28.0	0	14.0	28.0	55.9	65.9
6-146-4	C4689-1	0.12	0.56	82.2		10	6.68E+17	28.0	0	14.0	28.0	56.1	66.1
6-146-2	C4654-1	0.11	0.55	73.5		10	6.68E+17	25.1	0	12.5	25.1	50.1	60.1

Table 3-2b Adjusted Reference Temperatures for TPO 60-Year License (54 EFPY) – Unit 3 (continued)

Component	Heat	% Cu	% Ni	ChF	Adjusted ChF	Initial RT_{NDT} °F	1/4T Fluence n/cm^2	54 EFPY ΔRT_{NDT} °F	σ_I	σ_Δ	Margin °F	54 EFPY Shift °F	54 EFPY ART °F
Axial Welds:													
Lower Shell D1, D2, D3	37C065	0.182	0.181	94.5		-45	6.45E+17	31.7	16	15.8	45.0	76.7	31.7
Lower-Int Shell E1, E2, E3	37C065	0.182	0.181	94.5		-45	1.09E+18	41.0	16	20.5	52.0	93.0	48.0
Intermediate Shell F1, F2, F3	37C065	0.182	0.181	94.5		-45	6.68E+17	32.2	16	16.1	45.4	77.7	32.7
Circumferential Welds:													
Lower to Lower-Int DE	3P4000 Linde 124 Lot 3932	0.020	0.934	27.0		-50	6.45E+17	9.0	0	4.5	9.0	18.1	-31.9
Lower-Int to Intermediate EF	1P4217 Linde 124 Lot 3929	0.102	0.942	136.9		-50	6.68E+17	46.7	0	23.3	46.7	93.4	43.4
Nozzles:													
N16 [1]	C4689-1	0.12	0.56	82.2		10	3.16E+17	18.6	0	9.3	18.6	37.2	47.2
Best Estimate Chemistries from BWRVIP-135 R3:													
DE	3P4000	[[		27.0		-50	6.45E+17	9.0	0	4.5	9.0	18.1	-31.9
EF	1P4217		]]	139.3		-50	6.68E+17	47.5	0	23.8	47.5	95.0	45.0
Integrated Surveillance Program (BWRVIP-135 R3):													
Plate [2]	B0673-1	0.15	0.65	111.25		10	1.09E+18	48.3	0	17.0	34.0	82.3	92.3
Weld [3]	5P6756	0.06	0.93	82.0		-45	1.09E+18	35.6	0	17.8	35.6	71.1	26.1
Weld [3]	5P6756 [4]	[[	]]	108.0		-45	1.09E+18	46.9	0	23.4	46.9	93.7	48.7

Notes:

1. The N16 water level instrumentation nozzle occurs in the beltline region. Because the forging is fabricated from Alloy 600 material, the ART is calculated using the plate heats where the nozzles occur.
2. The ISP plate material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The ChF is determined using RG 1.99 for the ISP chemistry.
3. The ISP weld material is not the vessel target material and does not occur within the Unit 3 beltline region. Therefore, this material is not considered in determining the limiting ART. The ChF is determined using RG 1.99 for the ISP chemistry.
4. The ISP best estimate chemistry is used.

Table 3-3a Effects of Irradiation on RPV Circumferential Weld Properties for TPO 60-Year License (54 EFPY) – Unit 2

Parameter	NRC Staff Assessment for 64 EFPY [1] (Circumferential Welds)	PBAPS Unit 2 54 EFPY [2]
	(Chicago Bridge & Iron (CB&I) RPV)	(CB&I RPV)
Cu%	0.10	0.058
Ni%	0.99	0.949
ChF	134.9	79.2
Fluence at clad/weld interface (10^{19} n/cm^2)	1.02	0.124
$RT_{NDT(U)}$ (°F)	-65	-32
ΔRT_{NDT} w/o margin (°F) [Note 3]	135.6	36.5
Mean RT_{NDT} (°F)	70.6	4.5
P (F/E) NRC [Note 4]	1.78E-05	[Note 5]

Notes:

1. From Table 2.6-5 of Reference 27, with corrected ChF from Reference 31.
2. Best estimate chemistries are used for conservatism.
3. $\Delta RT_{NDT} = ChF * f^{(0.28 - 0.10 \log f)}$
4. P (F/E) stands for "Conditional Failure Probability".
5. Although a conditional failure probability has not been calculated, the fact that the PBAPS Unit 2 mean RT_{NDT} at the end of license is less than the 64 EFPY value provided by the NRC leads to the conclusion that the PBAPS Unit 2 RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL 98-05 (Reference 26).

Table 3-3b Effects of Irradiation on RPV Circumferential Weld Properties for TPO 60-Year License (54 EFPY) – Unit 3

Parameter	NRC Staff Assessment for 64 EFPY [1] (Circumferential Welds)	PBAPS Unit 3 54 EFPY [2]
	(CB&I RPV)	(CB&I RPV)
Cu%	0.10	0.104
Ni%	0.99	0.938
ChF	134.9	139.3
Fluence at clad/weld interface (10^{19} n/cm^2)	1.02	0.097
$RT_{NDT(U)}$ (°F)	-65	-50
ΔRT_{NDT} w/o margin (°F) [Note 3]	135.6	57.1
Mean RT_{NDT} (°F)	70.6	7.1
P (F/E) NRC [Note 4]	1.78E-05	[Note 5]

Notes:

1. From Table 2.6-5 of Reference 27, with corrected ChF from Reference 31.
2. Best estimate chemistries are used for conservatism.
3. $\Delta RT_{NDT} = ChF * f^{(0.28 - 0.10 \log f)}$
4. P (F/E) stands for “Conditional Failure Probability”.
5. Although a conditional failure probability has not been calculated, the fact that the PBAPS Unit 3 mean RT_{NDT} at the end of license is less than the 64 EFPY value provided by the NRC leads to the conclusion that the PBAPS Unit 3 RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL 98-05 (Reference 26).

Table 3-4a Effects of Irradiation on RPV Axial Weld Properties for TPO 60-Year License (54 EFPY) – Unit 2

Parameter	NRC BWRVIP-05 Supplement of SER (Mod 2) 32 EFPY	PBAPS Unit 2 54 EFPY [1]
Cu%	0.219	0.182
Ni%	0.996	0.181
ChF	231.1	94.5
Fluence at clad/weld interface (10^{19} n/cm^2)	0.148	0.165
$RT_{NDT(U)}$ (°F)	-2	-45
ΔRT_{NDT} w/o margin (°F) [Note 2]	116	49.6
Mean RT_{NDT} (°F)	114	4.6
Vessel Failure Frequency	5.02E-06	[Note 3]

Notes:

1. Beltline axial welds are the same heat (37C065). Bounding fluence is used for the analysis.
2. $\Delta RT_{NDT} = ChF * f^{(0.28 - 0.10 \log f)}$
3. Although a vessel failure frequency has not been calculated, the fact that the PBAPS Unit 2 mean RT_{NDT} at the end of license is less than the 32 EFPY values provided by the NRC leads to the conclusion that the PBAPS Unit 2 RPV vessel failure frequency is bounded by the NRC analysis, consistent with the requirements defined in BWRVIP-05 (Reference 28) and Reference 31.

Table 3-4b Effects of Irradiation on RPV Axial Weld Properties for TPO 60-Year License (54 EFPY) – Unit 3

Parameter	NRC BWRVIP-05 Supplement of SER (Mod 2) 32 EFPY	PBAPS Unit 3 54 EFPY [1]
Cu%	0.219	0.182
Ni%	0.996	0.181
ChF	231.1	94.5
Fluence at clad/weld interface (10^{19} n/cm^2)	0.148	0.157
$RT_{NDT(U)}$ (°F)	-2	-45
ΔRT_{NDT} w/o margin (°F) [Note 2]	116	48.5
Mean RT_{NDT} (°F)	114	3.5
Vessel Failure Frequency	5.02E-06	[Note 3]

Notes:

1. Beltline axial welds are the same heat (37C065). Bounding fluence is used for the analysis.
2. $\Delta RT_{NDT} = ChF * f^{(0.28 - 0.10 \log f)}$
3. Although a vessel failure frequency has not been calculated, the fact that the PBAPS Unit 3 mean RT_{NDT} at the end of license is less than the 32 EFPY values provided by the NRC leads to the conclusion that the PBAPS Unit 3 RPV vessel failure frequency is bounded by the NRC analysis, consistent with the requirements defined in BWRVIP-05 (Reference 28) and Reference 31.

Table 3-5 CUF and P+Q Stress Range of Limiting Components

	P + Q Stress (ksi)			CUF				
Component	**Current AOR (4,030 MWt) [2]**	**TPO (4,030 MWt) [4]**	**Allowable (ASME Code Limit)**	**Current AOR 40 years (4,030 MWt) [7]**	**Current AOR 60 years (4,030 MWt)**	**TPO 40 years (4,030 MWt) [4]**	**TPO 60 years (4,030 MWt) [4]**	**Allowable (ASME Code Limit)**
Shroud Support [1] (Attachment to RPV Location)	39.3 [3]	59.0 [5]	69.9 [6]	0.17	0.26	0.17 [6]	0.26 [6]	1.0

Notes:

1. The bounding stress value in the faulted condition for this component was revised due to a change in acoustic loads as a result of GEH SCs. The change was not due to implementation of TPO.
2. The current AOR was conservatively evaluated for 102% (per RG 1.49) of EPU and MELLLA+ (3,951 * 1.02 = 4,030 MWt).
3. The current AOR, which considered acoustic loads (faulted load case), conservatively used the design allowable (2.25 * Sm = 52.0 ksi) in the reconciliation evaluation. The allowable reported in the OLTP evaluation was the design allowable as it was based on design conditions and did not include acoustic loads. The referenced result pre-dates SC 12-20 and SC 13-08.
4. Consistent with Note 2, the TPO was conservatively evaluated for 4,030 MWt.
5. The change in value from AOR to TPO is due to including SCs, not implementation of TPO.
6. The TPO reconciliation evaluation (SC 12-20 and SC 13-08 evaluations) included acoustic loads in the load combination. Because acoustic loads are a faulted load, the ASME allowable increased to 3 * S_m = 69.9 ksi.
7. No change in CUF calculation due to inclusion of revised acoustic loads.

Table 3-6 Governing Stress Results for RPV Internals

Item	Component	Location	Service Level	Stress Category/Other	Unit	1.02 x CLTP Value(1)	Allowable Limit
1	Shroud	Top Guide Wedge	B	P_m+P_b	psi	18,070	21,450
2	Shroud Support(2)	Legs	B	P_m+P_b	psi	34,720	35,000
3	Core Plate	Longest Beam	B	ΔP	psi	27.34	31.64
		Core Plate Plug(3)	A	ΔP	psi	24.94	35.00
4	Top Guide	Longest Beam	B	P+Q	psi	34,200	50,700
5	CRD Housing	CRD Housing inside RPV	B	P_m	psi	13,090	16,185
6.a	CRGT	Tube	D	P_m	psi	11,408	16,000
6.b		Mid-span	B	Buckling Criteria	N/A	0.42	0.45
7	OFS	OFS Body	B	P_m+P_b	psi	6,682	15,580
8	Fuel Channel	Qualified by GNF (using a proprietary method)					

Table 3-6 Governing Stress Results for RPV Internals (continued)

Item	Component	Location	Service Level	Stress Category/Other	Unit	1.02 x CLTP Value(1)	Allowable Limit
9	Shroud Head and Separator Assembly (Including the Shroud Head Bolts)	N/A	A	N/A	Required Number of Bolts	32	≥ 32
10.a	Jet Pump Assembly	Riser Brace	D	P_m+P_b	psi	49,759	60,480
10.b		Beam Bolt	A	Load	lbs	15,212	17,442(4)
11	Access Hole Cover(5)	Bolt (Unit 3)	D	P_m+P_b	psi	147,481	154,000
		Plate (Unit 2)		P_m+P_b	psi	69,272	69,900
12.a	Core Spray Line(6)	Elbow	B	P_m+P_b	psi	15,370	20,920
12.b	Core Spray Sparger(6)	Tee Junction	B	P_m	psi	6,000	21,450
13	Feedwater Sparger	Header Pipe to Spray Nozzle Adaptor Weld	A, B	See Table 3-7 for fatigue usage factor			
14	In-Core Housing and Guide Tube	In-Core Housing at RPV Penetration	B	P_m	psi	< 20,270	24,900
15	Core Differential Pressure Line	Pipe	D	P_b	psi	29,340	49,950

Table 3-6 Governing Stress Results for RPV Internals (continued)

Item	Component	Location	Service Level	Stress Category/Other	Unit	1.02 x CLTP Value(1)	Allowable Limit
16	Jet Pump Instrument Penetration Seal	Safe End / Nozzle End	B	P+Q	psi	21,910	80,100

Notes:

1. TPO values are bounded by 1.02 x CLTP values. Stresses reported are for the limiting loading condition, with the least margin of safety. Normal (Level A) condition loads are bounded by that of the Upset (Level B) condition. Refer to MUR LAR Attachment 10 for the steam dryer evaluation.
2. The calculated stress for the shroud support is conservative.
3. PBAPS Unit 2 and 3 installed replacement extended core plate plugs.
4. The beam bolt load was qualified for 102% of CLTP and remains qualified for TPO condition. The 102% of CLTP and TPO allowable preload of 17,442 lbs accounts for the reduction in preload due to the 60-year fluence and operating temperature.
5. For Unit 3, the access hole cover bolt is more limiting; for Unit 2, the access hole cover plate is more limiting.
6. The repair of the core spray line and sparger was qualified for 102% of CLTP and remains qualified for TPO condition because all applicable loads at TPO conditions remain unchanged or bounded by the 102% of CLTP condition.

Table 3-7 Fatigue Usage Factors for RPV Internals for 60-Year Plant Life

Item	Component	1.02 x CLTP CUF (60-year Life)	TPO CUF (60-year Life)	Allowable
1	Shroud	0.89	0.89	1.0
2	Shroud Support	0.26	0.26	1.0
3	Core Plate	< 0.1	< 0.1	1.0
4	Top Guide	0.65	0.65	1.0
5	CRD Housing	Negligible (See Note 1)		1.0
6	CRGT	Negligible (See Note 2)		1.0
7	Orificed Fuel Support	Negligible (See Note 2)		1.0
8	Shroud Head and Separators Assembly (Include Shroud Head Bolts)	Negligible (See Note 3)		1.0
9	Jet Pump	0.22	0.22	1.0
10	Access Hole Cover	Negligible (See Note 3)		1.0
11a	Core Spray Line (4)	0.25	0.25	1.0
11b	Core Spray Sparger (4)	0.30	0.30	1.0
12	Feedwater Sparger	0.48	0.48	1.0
13	In-Core Housing and Guide Tube	Negligible (See Note 3)		1.0
14	Core Differential Pressure Line	Negligible (See Note 3)		1.0

Notes for Table 3-7:

1. The CRD housing is primarily subjected to mechanical loadings and the thermal/secondary stresses are small. The small value of the alternating stress (S_a) results in an infinite number of allowable fatigue cycles and a negligible fatigue usage factor.
2. The CRGT and OFS are primarily subject to mechanical loadings and thermal/secondary stresses are small. The small magnitude of the S_a results in an infinite number of allowable fatigue cycles.
3. The effect of temperature change on the thermal stress in the RPV to shroud annulus and the lower plenum is deemed to be small; hence, the fatigue usage factor is deemed to be negligible.
4. The repair of the core spray line and sparger was qualified for 102% of CLTP and remains qualified for TPO condition because all applicable loads at TPO conditions remain unchanged or bounded by the 102% of CLTP condition.

4.0 ENGINEERED SAFETY FEATURES

4.1 CONTAINMENT SYSTEM PERFORMANCE

TLTR Appendix G presents the methods, approach, and scope for the TPO uprate containment evaluation for LOCA. The existing plant-specific containment evaluations were performed at 102% of CLTP. Although the nominal operating conditions change slightly because of the TPO uprate, the required initial conditions for containment analysis inputs remain the same as previously documented in the current licensing basis which includes approved amendments for EPU (Reference 2) and MELLLA+ (Reference 3).

The following table summarizes the effect of the TPO uprate on the various aspects of the containment system performance, and was compared to the current evaluation performed at 102% of CLTP.

Topic	Key Parameters	TPO Effect
Short-Term Pressure and Temperature Response		Current analysis based on 102% of CLTP
Gas Temperature	Break Flow and Energy	
Pressure	Break Flow and Energy	
Long-Term Suppression Pool Temperature Response		
Bulk Pool	Decay Heat	
Local Temperature with SRV Discharge	Decay Heat	
Containment Dynamic Loads		
Loss-of-Coolant Accident Loads	Break Flow and Energy	
Safety-Relief Valve Loads	Decay Heat	
Sub-compartment Pressurization	Break Flow and Energy	
Containment Isolation Section 4.1.1 provides confirmation that motor-operated valves (MOVs) are capable of performing design basis functions at TPO conditions.		The ability of containment isolation valves and operators to perform their required functions is not affected because the evaluations have been performed at 102% of CLTP.

4.1.1 Generic Letter 89-10

The MOV requirements in the UFSAR were reviewed, and no changes to the functional requirements of the GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (Reference 38), MOVs, were identified as a result of operating at the TPO RTP level. The analyses of safety-related MOVs within the MOV program use maximum line pressures, maximum differential pressures, and maximum ambient temperatures that bound operation at TPO conditions. Therefore, the GL 89-10 MOVs remain capable of performing their design basis functions.

4.1.2 Generic Letter 96-05

GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Reference 39), was reviewed and determined to have no effects related to this TPO uprate.

4.1.3 Generic Letter 95-07 Program

The evaluation performed in support of GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (Reference 40), has been reviewed and no changes are identified as a result of operating at the TPO RTP level. The criteria for susceptibility to pressure locking or thermal binding were reviewed and it was determined that the slight changes in operating or environmental conditions expected to result from the TPO uprate would have no effect on the functioning of power-operated gate valves within the scope of GL 95-07. Therefore, the valves remain capable of performing their design basis functions.

4.1.4 Generic Letter 96-06

The PBAPS response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions" (Reference 41) was reviewed for the TPO uprate. The containment design temperatures and pressures in the current GL 96-06 evaluation are not exceeded during post-accident conditions for the TPO uprate. Therefore, the PBAPS response to GL 96-06 remains valid under TPO uprate conditions.

4.1.5 Generic Letter 89-16

GL 89-16, "Installation of a Hardened Wetwell Vent" (Reference 42), was reviewed and determined to have no effects related to this TPO uprate.

4.1.6 Containment Coatings

The normal heat loads inside containment are slightly increased; however, the required initial conditions for containment analysis inputs remain the same for the TPO uprate. The Service Level 1 coatings in the drywell have been determined to be acceptable to 340^{o}F, 70 psig, and $\geq 1x10^9$ rads. The maximum post-accident primary containment conditions do not change with the TPO uprate. Therefore, the containment coatings continue to bound the DBA temperature, pressure, and radiation at TPO conditions.

4.2 EMERGENCY CORE COOLING SYSTEMS

4.2.1 High Pressure Coolant Injection

The HPCI system is a steam driven high pressure injection system designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small break LOCA that does not immediately depressurize the reactor vessel.

The plant-specific evaluation specifically determined that there is no change in the following:

- Operating pressure;
- Pressure setpoints of the SRVs;
- The capability of the turbine-driven HPCI system to successfully develop the horsepower and speed required by the pumps;
- Startup capability of the turbine startup logic;
- HPCI capacity; and
- Decay heat calculations.

The TPO uprate does not affect the HPCI system operation, initiation, or capability requirements.

4.2.2 Core Spray

The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant makeup for a large break LOCA and for any small break LOCA after the RPV has depressurized. It also provides spray cooling for long-term core cooling in the event of a LOCA.

A plant-specific evaluation of the CS system was performed for PBAPS using the evaluation approach provided in TLTR Section 5.6.10 and Appendix J.2.3.1.

The plant-specific evaluation specifically determined that there is no change in the following:

- CS capacity; and
- Decay heat calculations.

The TPO uprate does not affect the CS system operation, initiation, or capability requirements.

4.2.3 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to provide reactor vessel coolant makeup during a large break LOCA or small break LOCA after the RPV has depressurized.

A plant-specific evaluation of the LPCI mode was performed for PBAPS using the evaluation approach provided in TLTR Section 5.6.4.

The plant-specific evaluation specifically determined that there is no change in the following:

- LPCI mode capacity; and
- Decay heat calculations.

The TPO uprate does not affect the LPCI mode of the RHR system operation, initiation, or capability requirements.

4.2.4 Automatic Depressurization System

The automatic depressurization system (ADS) uses SRVs to reduce the reactor pressure following a small break LOCA when it is assumed that the high pressure systems have failed. This allows CS and LPCI to inject coolant into the RPV. The ADS initiation logic and valve control is not affected by the TPO uprate.

The plant-specific evaluation specifically determined that:

- Pressure setpoints of the ADS valves are unchanged; and
- The ADS initiation logic and the ADS valve control are not affected by the TPO uprate operating conditions.

The plant-specific evaluation also concludes that the performance of the existing ADS valves remains unchanged because the current small-break LOCA analysis, performed at 102% of CLTP, bounds the TPO uprate conditions.

The TPO uprate does not affect the ADS system operation, initiation, or capability requirements.

4.2.5 ECCS Net Positive Suction Head

The most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. A plant-specific evaluation of the PBAPS containment was performed using the evaluation approach provided in TLTR Appendix G. All of the CLTP containment analyses were based on 102% of CLTP except for the Appendix R, station blackout (SBO) and ATWS events. These events have now been evaluated at the TPO bounding thermal power of 101.7%, or 4,018 MWt, which demonstrates that there continues to be positive NPSH margin with an acceptable time limit of operation in the zone of maximum erosion. Therefore, the TPO uprate does not affect compliance with the ECCS pump NPSH requirements. NPSH requirements continue to be met without reliance on containment accident pressure (CAP).

A conservative error was identified in the NPSH evaluations for Appendix R Cases A1, C1A and C1B at EPU conditions. These evaluations should have used a service water (SW) temperature of 86°F as indicated in Table 9.2f of the EPU LAR, but instead used 92°F, and therefore have been re-performed with the corrected temperature. The SW temperature of 92°F is a TS limit, while 86°F is a nominal value based on a statistical analysis of a five-year sampling of data for the months of June, July, August and September. This corrected evaluation provides increased NPSH margin for the EPU Appendix R cases. An extent of condition review performed during the TPO evaluation phase concluded that no other analyses were affected by this error. Table 4-1 provides the original EPU, corrected EPU and TPO values for the Appendix R cases.

It also includes the results of the maximum erosion zone evaluation indicating, in accordance with NRC draft guidance in Staff Requirements Memorandum (SRM) SECY 11-0014 (Reference 43), the time determined to be spent below a net positive suction head available (NPSHA) margin ration of 1.6. Table 4-2 provides the original EPU, corrected EPU and TPO values for NPSH margin.

The ATWS and SBO events were also re-analyzed for TPO conditions and, therefore, a plant-specific ECCS NPSH evaluation was performed for TPO for each event. The methodology used for the ECCS NPSH analysis is identical to that used in the PBAPS PUSAR (Reference 14). There is a decrease in ECCS NPSH margin for TPO over that of EPU due to a higher suppression pool temperature for TPO. The results of the ATWS, SBO, and limiting Appendix R Case C1B TPO NPSH analyses are shown in Tables 4-3, 4-4 and 4-5, respectively.

The RHR pumps have been analyzed for plant-specific conditions and have sufficient NPSH margin to perform satisfactorily during an ATWS initiated at TPO conditions. This plant-specific analysis is consistent with M+LTR SER Limitations and Conditions 12.17, 12.18.d, 12.23.1, 12.23.9, 12.23.10 and 12.24.4 (see Appendix B) concerning evaluation of the safety system performance during the long-term cooling phase of an ATWS in terms of available NPSH.

Therefore, PBAPS meets all M+LTR dispositions, as adjusted for TPO conditions, for the ECCS pump NPSH.

4.2.5.1 ECCS Suction Strainer Debris Loading

Consideration of ECCS suction strainer debris loading within the NPSH evaluations at TPO uprate conditions is consistent with the PBAPS current analysis-of-record for the design basis LOCA event. For the PBAPS TPO uprate, the small steam line break and ATWS events also include ECCS suction strainer debris loading in the NPSH evaluations for these events.

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The ECCS is designed to provide protection against a postulated LOCA caused by ruptures in the primary system piping. The current 10 CFR 50.46, or LOCA, analyses for PBAPS (Reference 44), which consider EPU (Reference 14) and MELLLA+ (Reference 15), have been performed at 102% of CLTP and therefore bound TPO uprate conditions, consistent with Appendix K. Table 4-6 shows the results of the PBAPS ECCS-LOCA analysis. The ECCS-LOCA results for PBAPS are in conformance with the licensing requirements of 10 CFR 50.46. Therefore, the CLTP LOCA analysis for GNF2 fuel bounds the TPO uprate for PBAPS.

Reference 45 provides justification for the GNF2 elimination of the 1,600°F upper bound PCT limit and justification that the licensing basis PCT will be conservative with respect to the upper bound PCT. The NRC SER for Reference 45 accepted this position, noting that because plant-specific upper bound PCT calculations have been performed for all plants, other means may be used to demonstrate compliance with the original SER requirements.

These other means are acceptable provided there are no significant changes to a plant's configuration that would invalidate the existing upper bound PCT calculations. Reference 45 provided justification for the elimination of the upper bound PCT limit for PBAPS.

For the TPO uprate there are no planned changes to the plant configuration that would invalidate the Reference 45 PBAPS LOCA evaluation.

The CLTP LOCA analysis for GNF2 fuel, which considers EPU (Reference 14) and MELLLA+ (Reference 15), is concluded to bound the TPO uprate for PBAPS.

4.4 MAIN CONTROL ROOM ATMOSPHERE CONTROL SYSTEM

The main control room atmosphere is not affected by the TPO uprate. Control room habitability following a postulated accident at TPO conditions is unchanged because the control room envelope/habitability systems have previously been evaluated for radiation release accident conditions at 102% of CLTP. Therefore, the system remains capable of performing its safety function at the TPO conditions.

4.5 STANDBY GAS TREATMENT SYSTEM

The SGTS minimizes the offsite and control room dose rates during venting and purging of the containment atmosphere under abnormal conditions. The current capacity of the SGTS was selected to maintain the secondary containment at a slightly negative pressure during such conditions. This capability is not changed by the TPO uprate conditions. The SGTS can accommodate DBA conditions at 102% of CLTP. Therefore, the system remains capable of performing its safety function for the TPO uprate condition.

4.6 PRIMARY CONTAINMENT LEAK RATE TEST PROGRAM AND CONTAINMENT ISOLATION SYSTEM

The PBAPS design includes the primary containment leak rate test (PCLRT) program and the containment isolation system. The PCLRT program is designed to enable testing of the primary containment isolation valves and the primary containment structure during non-operational conditions (i.e., systems tested while not in-service). Therefore, system operation is not affected by the TPO uprate.

The PCLRT program is not affected, because the reactor operating parameters are not changed for the TPO uprate and the current containment response analyses have been performed at 102% of CLTP. Based on no change in the post-accident short-term containment pressure and temperature, there is no revision necessary to the 10 CFR 50 Appendix J testing methodology and/or acceptance test criteria.

The containment isolation system is not affected by TPO uprate. The system uses setpoints developed to ensure containment isolation based on postulated accidents as expressed in the UFSAR considering the current licensing basis which includes approved amendments for EPU (Reference 2) and MELLLA+ (Reference 3). These setpoints utilize a 2% uncertainty factor required by RG 1.49. Because the TPO uprate reduces the RG 1.49 uncertainty from 2% to 0.34%, the previous analysis remains bounding.

Therefore, system operation is not affected by the TPO uprate.

4.7 POST-LOCA COMBUSTIBLE GAS CONTROL SYSTEM

The post-LOCA combustible gas control system was originally designed to maintain the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the flammability limit.

The revised 10 CFR 50.44 (68 FR 54123, dated September 16, 2003) does not define a design basis LOCA hydrogen release and eliminates the requirements for hydrogen control systems to mitigate such releases. PBAPS license amendment numbers 256 and 259 for Units 2 and 3 respectively, issued in 2005 (Reference 46), eliminated the requirements for the hydrogen/oxygen monitors and PBAPS license amendment numbers 274 and 278 for Units 2 and 3 respectively, issued in 2010 (Reference 47), eliminated the requirements for the containment atmospheric dilution system.

Table 4-1 ECCS Pump NPSHA Appendix R

Event	EPU Original (ft)	EPU Corrected (ft)	TPO (ft)	TPO Time <1.6 Margin Ratio (hrs)
Appendix R Case A1 RHR	17.21	18.93	17.70	12
Appendix R Case C1A RHR	20.51	22.37	21.86	18
Appendix R Case C1B RHR	16.03	16.88	16.28	16
Appendix R Case C1A CS	20.77	22.63	22.13	32

Table 4-2 ECCS Pump NPSH Margin Appendix R

Event	EPU Original (ft)	EPU Corrected (ft)	TPO (ft)
Appendix R Case A1 RHR	1.21	2.93	1.70
Appendix R Case C1A RHR	4.51	6.37	5.86
Appendix R Case C1B RHR	0.03	0.88	0.28
Appendix R Case C1A CS	0.77	2.63	2.13

Table 4-3 RHR Pump NPSH ATWS Event

Parameter	Units	Current Licensing Basis	TPO
Maximum Torus Temperature	°F	171.1	171.7
Net Positive Suction Head Required Effective ($NPSHR_{eff}$)	feet	16.00	16.00
NPSHA	feet	30.76	30.57
NPSH Margin	feet	14.76	14.57

Table 4-4 RHR Pump NPSH SBO Event

Parameter	Units	Current Licensing Basis	TPO
Maximum Torus Temperature	°F	199.0	200.0
$NPSHR_{eff}$	feet	16.00	16.00
NPSHA	feet	20.75	20.18
NPSH Margin	feet	4.75	4.18

Table 4-5 RHR Pump NPSH Appendix R Case C1B Event

Parameter	Units	Current Licensing Basis[1]	TPO
Maximum Torus Temperature	°F	203	204
$NPSHR_{eff}$	feet	16.00	16.00
NPSHA	feet	16.88	16.28
NPSH Margin	feet	0.88	0.28

Note 1: These values reflect the corrected calculation with a service water temperature of 86°F.

Table 4-6 PBAPS ECCS-LOCA Analysis Results for GNF2 Fuel

Parameter	Current AOR[1]	10 CFR 50.46 ECCS-LOCA Analysis Acceptance Criteria
Nominal PCT	1,551°F	N/A
Appendix K PCT	1,910°F	≤ 2,200°F
Licensing Basis PCT	<1,925°F	≤ 2,200°F
Maximum Local Oxidation	<4.0%	≤ 17%
Core-Wide Metal-Water Reaction	<0.1%	≤ 1.0%

Note 1: The current AOR bounds TPO conditions.

5.0 INSTRUMENTATION AND CONTROL

5.1 NSSS MONITORING AND CONTROL

The instruments and controls that directly interact with or control the reactor are usually considered within the NSSS. The NSSS process variables and instrument setpoints that could be affected by the TPO uprate were evaluated.

5.1.1 Neutron Monitoring System

5.1.1.1 Average Power Range Monitors and Wide Range Neutron Monitors

The average power range monitors (APRMs) are re-calibrated to indicate 100% at the TPO RTP level of 4,016 MWt. The APRM high flux scram and the upper limit (clamp) of the flow-biased scram and rod block setpoints, expressed in units of percent of licensed power, are not changed. The flow-biased APRM trips, expressed in units of absolute thermal power (i.e., MWt), remain the same. Thus, the MCPR reduction or maximum LHGR ratio to the limiting value is unchanged for potential transient increases of power from the operating limit to the APRM rod block alarm or flow-referenced scram trip.

For the TPO uprate, no adjustment is needed to ensure the wide range neutron monitors (WRNMs) have adequate overlap with the APRMs. However, normal plant surveillance procedures may be used to adjust the overlap of the WRNMs with the APRMs. The WRNM channels' short reactor period scram and rod block trips are unchanged for the TPO uprate.

5.1.1.2 Local Power Range Monitors and Traversing In-Core Probes

At the TPO RTP level, the flux at some LPRMs increases. However, the small change in the power level is not a significant factor to the neutronic service life of the LPRM detectors and the radiation level of the traversing in-core probes (TIPs). It does not change the number of cycles in the lifetime of any of the detectors. The LPRM accuracy at the increased flux is within specified limits, and the LPRMs are designed as replaceable components. The TIPs are stored in shielded rooms. The radiation protection program for normal plant operation can accommodate a small increase in radiation levels.

In accordance with Methods LTR SER Limitation and Condition 9.17 (Reference 12) and M+LTR SER Limitation and Condition 12.15 (Reference 11) the predicted bypass void fraction at the D-Level LPRMs satisfies the [[ ]] design requirement for PBAPS MELLLA+ and TPO. The SRLR will validate that the power distribution in the core is achieved while maintaining individual fuel bundles within the allowable thermal limits as defined in the COLR. When moving down and left on the MELLLA+ upper boundary, the hot channel exit void in the bypass region increases. The hot channel exit void in the bypass region does not exceed [[ ]] in the MELLLA+ operating domain (including the expanded TPO region) as shown in Table 5-1.

Because thermal neutron TIPs are affected by bypass voiding above the D-level LPRMs in excess of [[ ]], operator actions and procedures that mitigate the effect of bypass voiding on the thermal TIPs and the core simulator used to monitor the fuel performance are requested in M+LTR SER Limitation and Condition 12.15 for operation. These items are not required for PBAPS because of the use of gamma TIPs and because hot channel bypass voiding at the TIP exit elevation is not in excess of [[ ]] for the entire MELLLA+ operating domain (including the expanded TPO region) as shown in Table 5-1.

5.1.1.3 Rod Block Monitor

The RBM instrumentation is referenced to an APRM channel. Because the APRM has been rescaled, there is only a small effect on the RBM performance due to the LPRM performance at the higher average local flux. The RBM instrumentation is not significantly affected by the TPO uprate conditions, and no change is needed.

5.1.2 Rod Worth Minimizer

The rod worth minimizer (RWM) does not perform a safety-related function. The function of the RWM is to support the operator by enforcing rod patterns until reactor power has reached appropriate levels. The power-dependent setpoints for the RWM are discussed in Section 5.3.8.

5.2 BOP MONITORING AND CONTROL

Operation of the plant at the TPO RTP level has a minimal effect on the BOP system instrumentation and control devices. The improved FW flow measurement, which is the basis for the reduction in power uncertainty, is addressed in Section 1.4. Some BOP instruments will be re-scaled to expand the calibration range or will be replaced.

5.2.1 Pressure Control System

The pressure control system (PCS) provides a fast and stable response to steam flow changes so that reactor pressure is controlled within allowable values (AVs). The PCS consists of the pressure regulation system, turbine control valve system and steam bypass valve system. The main turbine speed/load control function is performed by the main turbine-generator electrohydraulic control (EHC) system. The steam bypass valve pressure control function is performed by the turbine bypass control system (TBCS).

An analysis of the PCS was performed at the TPO bounding power of 4,018 MWt. Based upon that analysis, no modification is required for the pressure regulation system or steam bypass valve system for TPO. No modifications are required for the operator interface indications, controls, or alarm annunciators provided in the main control room. The required adjustments are limited to tuning of the control settings that may be required for optimal operation.

The difference between the analyzed steam flow capabilities of the TCVs valves wide open (VWO) condition and an operating condition is the effective throttle flow margin. Satisfactory reactor pressure control by the turbine pressure regulator and the TCVs requires an adequate effective throttle flow margin. Existing operating procedures specify a limit for the position of the TCVs during normal operation. This constraint on TCV position will limit reactor power and steam flows to provide adequate effective throttle flow margin. The resulting effective throttle flow margin provides stable TCV positioning and ensures stable reactor pressure control and safe operation of the plant.

At full TPO RTP, the TCVs are predicted to operate close to, or beyond, the procedurally-controlled position limit. The procedural limit on TCV position could prevent taking full advantage of the TPO power uprate. Any such potential power limitation would be an economic consideration with no effect on the safe operation of the plant. Subsequent to this TPO uprate, turbine-related hardware may be modified to take full advantage of the TPO power uprate. Such modification would be processed in accordance with 10 CFR 50.59.

PCS tests, consistent with the guidelines in TLTR Appendix L, will be performed during the power ascension phase.

5.2.2 EHC Turbine Control System

The PCS is discussed in Section 5.2.1.

While no modifications are required for TPO, PBAPS is replacing the existing analog EHC system with digital EHC control implemented in accordance with the requirements of 10 CFR 50.59. The new digital EHC system contains similar functions of control as the current analog system. Both the analog EHC system and the digital EHC system have been reviewed and found to be adequate for TPO conditions.

5.2.3 Feedwater Control System

An evaluation of the ability of the FW control system, turbine driven FW pump control valves, and FW turbine controls to maintain adequate water level control at the TPO uprate conditions has been performed. The 1.9% increase in FW flow associated with the bounding TPO uprate power is within the current control margin of these systems. No changes in the operating reactor water level or reactor water level trip setpoints are required for the TPO uprate. FW control system tests, consistent with the guidelines in TLTR Appendix L, will be performed during the power ascension phase.

5.2.4 Leak Detection System

The setpoints associated with leak detection have been evaluated with respect to the ~2% higher steam flow and ~2°F increase in FW temperature for the TPO uprate. Each of the systems where leak detection could be potentially affected is addressed below.

Main Steam Tunnel Temperature Based Leak Detection

The ~2°F increase in FW temperature for the TPO uprate decreases the leak detection trip avoidance margin. The setpoints for initiation of MSIV closure on high steam tunnel temperature remain unchanged because steam line temperature is unchanged (constant vessel dome pressure), and the increase in FW temperature is very small. There is no significant loss of margin for trip avoidance compared to CLTP operation. The current setpoints maintain the safety functions within the current design and licensing bases. Thus, because there is no loss of protection or significant loss of margin for trip avoidance compared to CLTP operation, the high steam tunnel temperature setpoint remains unchanged.

RWCU System Temperature Based Leak Detection

There is no significant effect on RWCU system temperature or pressure due to the TPO uprate. Therefore, there is no effect on the RWCU temperature based leak detection.

HPCI System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the HPCI system temperature or pressure, and thus, the HPCI temperature based leak detection system is not affected.

RCIC System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the RCIC system temperature or pressure, and thus, the RCIC temperature based leak detection system is not affected.

RHR System Temperature Based Leak Detection

The TPO uprate does not increase the nominal vessel dome pressure or temperature. Therefore, there is no change to the RHR system temperature or pressure, and thus, the RHR temperature based leak detection system is not affected.

Non-Temperature Based Leak Detection

The non-temperature based leak detection systems are not affected by the TPO uprate.

5.3 TECHNICAL SPECIFICATION INSTRUMENT SETPOINTS

The determination of instrument setpoints is based on plant operating experience, conservative licensing analyses or limiting design/operating values. Standard GEH setpoint methodologies (References 6 and 48*)* are used to generate the AVs and nominal trip setpoints (NTSPs) related to any AL change, as applicable. Each actual trip setting is established to preclude inadvertent initiation of the protective action, while assuring adequate allowances for instrument accuracy, calibration, drift and applicable normal and accident design basis events.

Table 5-2 lists the ALs, or DLs if no ALs, that change based on results from the TPO evaluations and safety analyses. In general, if the AL does not change in the units shown in the TS, then no change in its associated plant AV and NTSP is required, as shown in the TS. Changes in the setpoint margins due to changes in instrument accuracy and calibration errors caused by the change in environmental conditions around the instrument due to the TPO uprate are negligible. Maintaining constant nominal dome pressure for the TPO uprate minimizes the potential effect on these instruments by maintaining the same fluid properties at the instruments. The setpoint evaluations are based on the guidelines in TLTR (Reference 1) Section 5.8 and Appendix F, Section 4; and on Section 5.3 of Reference 6.

5.3.1 High Pressure Scram

The high pressure scram terminates a pressure increase transient not terminated by direct or high flux scram. Because there is no increase in nominal reactor operating pressure with the TPO uprate, the scram AL on reactor high pressure is unchanged.

5.3.2 Hydraulic Pressure Scram and Recirculation Pump Trip

The AL for the turbine hydraulic pressure (low oil pressure trip) that initiates the T/G trip scram and EOC RPT at high power remains the same as for CLTP. No modifications are being made to the turbine hydraulic control systems for TPO; actuation of these safety functions remains unchanged for TPO.

5.3.3 High Pressure Recirculation Pump Trip

The ATWS-RPT trips the pumps during plant transients with increases in reactor vessel dome pressure. The ATWS-RPT provides negative reactivity by reducing CF during the initial part of an ATWS. The evaluation in Section 9.3.1 demonstrates that the TS limit for the high pressure ATWS-RPT is acceptable for the TPO uprate.

5.3.4 Safety Relief Valve

Because there is no increase in reactor operating dome pressure, the SRV ALs are not changed.

5.3.5 Main Steam Line High Flow Isolation

The TS AV of this function is expressed in terms of psid. Although the MS flow increases by ~2%, the MSL high steam flow AL in terms of differential pressure is not changed for the TPO uprate. The corresponding AL in terms of steam flow is decreased as the result of higher absolute flow at TPO. Because of the large spurious trip margin, sufficient margin to the trip setpoint exists to allow for normal plant testing of the MSIVs.

5.3.6 Fixed APRM Scram

The fixed APRM ALs expressed in percent of LTP do not change for the TPO uprate. The guidelines presented in TLTR Section F.4.2.2 are applicable to PBAPS. A plant-specific evaluation of the fixed APRM ALs was performed for PBAPS using the evaluation approach provided in TLTR Section F.4.2.2.

The plant-specific evaluation specifically verified that the limiting transient that relies on the fixed APRM trip is the vessel overpressure transient (MSIVC) with indirect scram. This event analysis accounts for 102% of CLTP and is reanalyzed on a cycle specific basis.

The plant-specific evaluation concludes that confirmation analysis at the time of the first reload is sufficient to confirm that the upper limits of the APRM trip and alarm setpoints expressed in units of percent of licensed power will not change.

5.3.7 APRM Simulated Thermal Power Scram and Rod Block Functions

The simulated thermal power (STP) APRM DLs, for both TLO and SLO, are unchanged in units of absolute core thermal power versus recirculation drive flow. Because the setpoints are expressed in percent of LTP, they decrease in proportion to the power uprate or TPO RTP. This is the same approach taken for generic BWR uprates described in ELTR1 (Reference 4). There is no significant effect on the instrument errors or uncertainties from the TPO uprate. Therefore, the AV and NTSP are established by directly incorporating the change in the DL.

5.3.8 Rod Worth Minimizer Low Power Setpoint

The RWM low power setpoint (LPSP) is used to enforce the rod patterns established for the control rod drop accident at low power levels. The RWM LPSP AL of 10% of LTP is not changed by TPO. It is conservative to keep the existing percent of rated power after TPO uprate. The guidelines in TLTR Section F.4.2.9 are applicable to PBAPS.

5.3.9 Rod Block Monitor

The severity of the rod withdrawal error (RWE) during power operation event is dependent upon the RBM rod block setpoint. [[

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5.3.10 Flow-Biased Rod Block Monitor

PBAPS does not have a flow-biased RBM system.

5.3.11 Main Steam Line High Radiation Isolation

Deleted per License Amendment 299 (Unit 2) and Amendment 302 (Unit 3) (Reference 49).

5.3.12 Low Steam Line Pressure MSIVC (RUN Mode)

The purpose of this function is to initiate MSIVC on low steam line pressure when the reactor is in the RUN mode. The change in steam line pressure near the turbine (where this sensor is located) will decrease slightly due to the higher steam flow, but will not change significantly compared to the nonlimiting nature of the Pressure Regulator Failure (Open) transient, which uses this function to mitigate the event. Its backup function for LOCA events is also maintained satisfactorily with the unchanged setpoint. Thus, the low steam line pressure setpoint for initiation of MSIVC in the RUN mode will be maintained at its current value for the TPO uprate.

5.3.13 Reactor Water Level Instruments

As described in TLTR Section F.4.2.10, the TPO uprate does not result in a significant increase in the possibility of a reactor scram, equipment trip, or ECCS actuation. Use of the current ALs maintains acceptable safety system performance. The low reactor water level ALs for scram and ADS/ECCS are not changed for the TPO uprate. The high water level ALs for trip of the main turbine and the FW pumps, and reactor scram, are not changed for the TPO uprate.

The water level change during operational transients is slightly affected by the TPO uprate. The plant response following the trip of one FW pump does not change significantly, because the maximum operating rod line is not being increased. Therefore, the final power level following a single FW pump trip at TPO uprate conditions would not change relative to the remaining FW flow as exists at CLTP.

5.3.14 Main Steam Line Tunnel High Temperature Isolations

As noted in Section 5.2.4, the high steam tunnel temperature setpoint remains unchanged for the TPO uprate.

5.3.15 Low Condenser Vacuum

In order to produce more electrical power, the amount of heat discharged to the main condenser increases slightly. This added heat load may slightly increase condenser backpressure, but the increase would be insignificant (< 0.15 in. HgA). The slight change in condenser vacuum after implementation of TPO will not adversely affect any trip signals associated with low condenser vacuum.

5.3.16 TSV Closure Scram, TCV Fast Closure Scram, and EOC-RPT Bypasses

The turbine first-stage pressure (TFSP) bypass allows the TSV closure scram and TCV fast closure scram to be bypassed, and the EOC-RPT bypass allows the EOC-RPT to be bypassed, when reactor power is sufficiently low, such that the scram and EOC-RPT functions are not needed to mitigate a T/G trip. This power level is the AL for determining the actual trip setpoint, which comes from the TFSP. The TFSP setpoint is chosen to allow operational margin so that scrams can be avoided, by transferring steam to the turbine bypass system during T/G trips at low power.

Based on the guidelines in TLTR Section F.4.2.3, the TSV closure scram, TCV fast closure scram, and EOC-RPT bypass AL in percent of LTP is reduced by the ratio of the power increase. The new AL does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant startup is maintained.

No modifications are made to the PBAPS turbine for the TPO uprate, so there is no change in the first-stage pressure/steam flow relationship from previous operation.

5.3.17 Locations in Technical Specifications where Percentage of RTP is Unchanged

The required changes to the TS and TS Bases to implement TPO are provided in MUR LAR Attachments 2 and 3. Unless specifically addressed in MUR LAR Attachments 2 and 3, no values of “% RTP” are changed.

Table 5-1 Hot Channel Bypass Voiding at Steady-State and Off-Rated Conditions

Statepoint on Power / Flow Map	Core Power (% of TPO RTP)	Core Flow (% of rated)	Hot Channel Void Fraction in Bypass Region at Core Exit (ISCOR)	Hot Channel Void Fraction in Bypass Region at TIP Exit (ISCOR Nodes 22 and 23 Average)	Hot Channel Void Fraction in Bypass Region at Instrumentation D-level (ISCOR Node 21)
D	100.0	101.5	0.000	0.000	0.000
J	100.0	85.2	0.000	0.000	0.000
K	77.5	55.0	0.016	0.000	0.000
L	67.3	55.0	0.013	0.000	0.000

Table 5-2 Analytical Limits and Design Limits for Current and TPO Power Level

Parameter	Current	TPO	Justification
APRM High Neutron Flux–Scram-Fixed (% LTP), AL	125	No change	
APRM Flow Biased (FB) STP – (Scram) (2)			(1)
FB STP Scram Clamp (% LTP) (4), DL	120	No change	
TLO FB STP Scram (% LTP) (3), DL	0.61W + 69.3	0.60W + 68.1	(4)
SLO FB STP Scram (% LTP) (3), DL	0.55W + 62.2	0.54W + 61.1	(4)
APRM FB STP (Rod Block) (2)			(1)
FB STP Rod Block Clamp (% LTP) (4), DL	110.4	No change	
TLO FB STP Rod Block (% LTP) (3), DL	0.61W + 59.7	0.60W + 58.7	(4)
SLO FB STP Rod Block (% LTP) (3), DL	0.55W + 52.6	0.54W + 51.7	(4)
TSV and TCV Closure Scram Bypass (% LTP), AL	26.7	26.3	(5)
MSL High Flow Isolation, ALs			
% rated steam flow	140.0	137.4	(5)
psid	179.23	No change	
Rod Worth Minimizer LPSP (% LTP), DL	10	No change	(1), (5)

Notes:

(1) PBAPS does not have ALs for these setpoint functions.

(2) No credit is taken in any safety analysis for flow-biased setpoints.

(3) W is % recirculation drive flow where 100% drive flow is that required to achieve 100% CF at 100% power.

(4) These changes to the DLs are based upon the methodology approved by the NRC in Reference 1.

(5) Limits scaled for an uprate of approximately 1.66% thermal power.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 AC Power

The plant electrical equipment ratings are given in Table 6-1.

A detailed comparison of existing ratings with ratings at TPO conditions and the effect of the TPO uprate on the main generator, main transformers, normal auxiliary transformers, and startup auxiliary transformer are shown in Tables 6-2, 6-3, 6-4, and 6-5, respectively. Operation at the TPO uprate conditions is not expected to have any effect on the operation of the backup auxiliary transformer.

6.1.1 Off-Site Power

The main generator, main transformer and isolated phase bus nameplate ratings are listed in Table 6-1 and discussed below:

- Main Generator: The Unit 2 generator is a direct-driven 3-phase 60 Hz, 22,000 V, 1,800 rpm, hydrogen inner-cooled, synchronous generator rated for 1,530 megavolt amps (MVA) at a 0.92 power factor (PF), with a 0.510 short circuit ratio at a nominal hydrogen pressure of 75 psig. The Unit 3 generator is a direct-driven 3-phase 60 Hz, 22,000 V, 1,800 rpm, hydrogen inner-cooled, synchronous generator rated for 1,530 MVA at a 0.90 PF, with a 0.540 short circuit ratio at a nominal hydrogen pressure of 75 psig.
- Main Transformer: The 1,530 MVA main power transformer consists of three single-phase, 510 MVA 22 – 539.5 Grd. Y/ 311.5 kV, oil directed, air forced, 65°C rise, 60 Hz, oil-filled type, outdoor transformers.
- Isolated Phase Bus Duct: The isolated phase bus duct consists of a main bus, a delta bus, and an auxiliary bus. The isolated phase bus continuous current rating is based on a 105°C operating temperature (65°C rise above a 40°C ambient temperature) with forced air cooling for the main bus and the delta bus, and self-cooling for the auxiliary buses. The main bus is rated at 42,300 A with a momentary fault current rating of 440,000 A. The delta bus is rated at 20,500 A with a momentary fault current rating of 440,000 A. The auxiliary bus subsections are rated at 2,000 A with a momentary fault current rating of 650,000 A. The voltage rating of the system is 25,000 V. The forced cooling is handled by an air handling unit with a design heat transfer capacity of 2,570,800 Btu/hr.

The review of the existing off-site electrical equipment concluded the following:

- The main generator will be operated within the existing generating capability curve for the TPO uprate. For summer and winter operations, the gross generator MWe output will be kept on, or within, the existing generator reactive capability curve.
- The isolated phase bus duct is adequate for both rated voltage and low voltage current output. The isolated phase bus duct cooling system capacity is adequate for the expected heat rejection loads during the TPO uprate operation. Therefore, the isolated phase bus duct cooling system is adequate to support the TPO uprate.

- The main transformers and the associated switchyard components (rated for maximum generator output) are adequate for the TPO uprate-related transformer output. The items with the least margin are the Unit 2 disconnect switches which have 25.6% margin.

A grid stability study and a voltage analysis for the TPO uprate are provided as MUR LAR Attachment 13.

The grid stability study considers the increase in electrical output to demonstrate conformance to General Design Criterion (GDC) 17 (10 CFR 50, Appendix A). GDC 17 addresses on-site and off-site electrical supply and distribution systems for safety-related components. There is no significant effect on grid stability or reliability. There are no modifications associated with the TPO uprate which would increase electrical loads beyond those levels previously included or which would require revising the logic of the distribution systems.

On the basis of the PECO voltage analysis for the TPO uprate, the capability of the transmission system to maintain the post-trip voltage drops and voltages at the safety buses above the reset value of the degraded voltage relay on a steady-state basis has been verified.

6.1.2 On-Site Power

The on-site power distribution system consists of transformers, numerous buses, and switchgear. Alternating current (AC) power to the distribution system is provided from the transmission system or from onsite diesel generators. The on-site distribution system loads were reviewed under normal and emergency operating scenarios. In both cases, the loads are computed based primarily on equipment nameplate data or brake horsepower (BHP). These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operation at the TPO RTP level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running BHP. Therefore, there are negligible changes to the load, voltage drop or short circuit current values.

The only identifiable change in electrical load demand is associated with the condensate and the reactor recirculation pumps. The new operating power requirements for the condensate pumps for TPO uprate are within the values evaluated for EPU and found acceptable. Reactor recirculation pumps horsepower requirements increase slightly (~0.08%) due to the TPO uprate conditions. Accordingly, there are negligible changes in the on-site distribution system design basis loads or voltages due to the TPO conditions. The system environmental design bases are unchanged. Operation at the TPO RTP level is achieved by utilizing existing equipment operating at or below the nameplate rating; therefore, under normal conditions, the electrical supply and distribution components (e.g., switchgear, motor control centers, and cables) are adequate.

Station loads under emergency operation and distribution conditions (emergency diesel generators (EDGs)) are based on operational requirements. The ECCS pump loading is based on station UFSAR design basis requirements. Emergency operation at the TPO uprate RTP levels is achieved by utilizing existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

ECCS loads were evaluated for EPU at 102% of CLTP, which bounds the TPO uprate. Therefore, the amount of power required to perform safety-related functions (pump and valve loads) does not increase, and the current emergency power system remains adequate. The systems have sufficient capacity to support all required loads for safe shutdown, to maintain a safe shutdown condition, and to operate the engineered safety feature equipment following postulated accidents.

Because the duty cycle and duration for design basis EDG loads is based on analytical power levels of at least 102% of CLTP, these loads will remain unchanged by the TPO uprate. Hence, the required reserve volume of emergency fuel oil is not changed. Therefore, usable emergency fuel oil reserves will be adequate to support the TPO uprate.

6.1.3 Emergency Diesel Generator

There are no modifications associated with the TPO uprate that would increase the electrical loads associated with the engineered safeguard and selected non-safeguard systems or alter the EDG subsystems. Therefore, the performance of the EDG and the 4kV emergency system is not affected by the TPO uprate.

6.2 DC POWER

The changes to the auxiliary power system as a result of the TPO uprate are small increases in the horsepower of the condensate pump and the reactor recirculation pump motors. The direct current (DC) system does not power the affected pumps; therefore, the DC system is not affected by the increase in motor duty. The DC system supplies power for control and auxiliary systems of the main equipment.

There are no changes to the DC system loading resulting from TPO. Thus, there is no effect on the DC system as a result of TPO.

The LEFM is operational in the plant and described in Section 7.10.3.3, Feedwater Flow Measurement, of the UFSAR.

6.3 FUEL POOL

The following sections address the fuel pool cooling and cleanup system (FPCCS), crud and corrosion products in the fuel pool, radiation levels and structural adequacy of the fuel racks. The changes due to TPO are within the DLs of the system and its components. The FPCCS meets the UFSAR requirements at the TPO conditions, which includes EPU (Reference 14) and MELLLA+ (Reference 15).

6.3.1 Fuel Pool Cooling

The spent fuel pool (SFP) heat load remains within the capability of the FPCCS as ensured by cycle-specific calculations to verify heat load is less than or equal to that previously analyzed. The TPO uprate does not affect the heat removal capability of the FPCCS supplemented with RHR assist mode, as shown in Table 6-6. The TPO heat load is within the design basis heat load for the FPCCS supplemented with RHR assist mode.

The SFP cooling and makeup adequacy is maintained by controlling the timing of the discharge (fuel offload) to the SFP to ensure the capability of the FPCCS to maintain adequate fuel pool cooling for the TPO uprate.

The FPCCS evaluation for CLTP conditions has been re-performed using a SFP volume of 37,439 ft^3 instead of 53,350 ft^3. The latter value was used in the original CLTP evaluation. However, upon review in preparation for the TPO evaluation, it was determined that the original evaluation SFP volume did not consider the volumes of the other components in the SFP (e.g., fuel racks, control rod blade racks) as displacing water in the pool. This corrected evaluation causes a decrease in the time to boil margins for the cases considered. The plant-specific evaluation, which was performed at 102% of CLTP, bounds the TPO uprate operating conditions.

Table 6-7 summarizes the three conservative, bounding SFP cooling evaluations for both CLTP and CLTP x 1.02 conditions: normal offload with full FPCCS capability, full-core offload with RHR cooling, and a normal offload with a single failure in the FPCCS. The predicted boil-off rates remain within the available makeup capability. The worst-case makeup requirement occurs when all cooling is lost after a full-core offload. If this condition occurs, refueling water, demineralized water, and condensate can each be aligned to provide sufficient makeup to maintain SFP level within one hour using only valve and pump manipulations. Other lower capacity systems are available within one hour, and other high capacity systems are available after one hour. The heating rate is sufficiently slow to allow operator actions to initiate makeup prior to the SFP reaching boiling.

The FPCCS heat exchangers are sufficient to remove the decay heat during normal refueling. The equipment required is not affected by TPO.

For a full core off-load, the RHR system in FPCCS assist mode is available to maintain the SFP water temperature below the DL.

6.3.2 Crud Activity and Corrosion Products

The crud activity and corrosion products associated with spent fuel can increase very slightly due to the TPO. The increase is insignificant and SFP water quality is maintained by the FPCCS.

6.3.3 Radiation Levels

The normal radiation levels around the SFP may increase slightly during fuel handling operation. This increase is acceptable and does not significantly increase the operational doses to personnel or equipment.

6.3.4 Fuel Racks

There is no effect on the design of the fuel racks because the maximum allowable spent fuel temperature is not being increased.

6.4 WATER SYSTEMS

The safety-related and non-safety-related cooling water loads potentially affected by TPO are addressed in the following sections. The environmental effects of TPO are controlled such that none of the present limits (e.g., maximum allowed cooling water discharge temperature) are increased.

Cooling water systems including SW, emergency service water (ESW), high pressure service water (HPSW), Turbine Building closed cooling water (TBCCW), Reactor Building closed cooling water (RBCCW), and chilled water (CW) were analyzed for 102% of CLTP. Therefore, these systems are acceptable for TPO uprate conditions.

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Loads

The safety-related SW systems (i.e., the ESW and HPSW systems) provide cooling water during and following design basis events. The performance of the safety-related SW systems during and following design basis events does not change because the current analyses were performed at 102% of CLTP. HPSW heat loads, when aligned to the emergency cooling tower (ECT), were evaluated at 102% of CLTP for the TPO uprate. Heat loads are within the existing capacity of the RHR and associated safety-related SW systems.

6.4.1.2 Non-Safety-Related Loads

The temperature of SW discharge results from the heat rejected to the SW system via closed cooling water systems and other auxiliary heat loads. The major SW heat load increases from the TPO reflect an increase in main generator losses rejected to the stator water coolers and hydrogen coolers and the TBCCW system. The increase in SW heat loads from these sources is approximately proportional to the power increase. Because the current SW analysis includes a 2% margin, it bounds the TPO uprate.

For normal operation, the analyzed discharge temperature of the SW system does not increase with TPO as the previous heat load and temperature analysis bounds TPO conditions. Therefore, the SW system is adequate for the TPO conditions.

6.4.2 Main Condenser/Circulating Water/Normal Heat Sink Performance

The main condenser, circulating water, and normal heat sink systems are designed to remove the heat rejected to the condenser and thereby maintain adequately low condenser pressure as recommended by the turbine vendor. TPO operation increases the heat rejected to the condenser and may reduce the difference between the operating pressure and the minimum condenser vacuum. The performance of the main condenser was evaluated for operation at TPO conditions. The evaluation confirms that the condenser, circulating water system and normal heat sink are adequate for TPO operation.

6.4.2.1 Discharge Limits

The Pennsylvania Department of Environmental Quality National Pollutant Discharge Elimination System (NPDES) permit provides the effluent limitations and monitoring requirements for discharging wastewater at the site. The TPO power uprate will not change chemical discharges controlled by the NPDES permit as no flow rate or chemical changes are being made for the TPO uprate. The increase in thermal discharge into the Conowingo Pond from the implementation of the TPO is expected to be less than 0.4°F. This slight increase in discharge temperature is not expected to have any significant effect on biological species in the Conowingo Pond. This statement is supported by results of detailed studies performed for potential biological effects in the Conowingo Pond during extreme ambient conditions under EPU conditions.

Therefore, no significant change in the types or amounts of effluents released into the environment will occur due to the TPO power uprate. Frequent monitoring of thermal discharges at the plant required to be performed under the current NPDES permit ensures that thermal permit limits are not exceeded. Cooling tower operation required under the current NPDES permit during warm weather months also provides appropriate protection of biology in the Conowingo Pond.

6.4.3 Chilled Water System

The CW system consists of the non-safety-related drywell chilled water system (DCWS), and the non-safety-related control room chilled water system (CRCWS). The heat load to the DCWS is not significantly affected by TPO (increase of ~0.1%) and the heat load will remain within the system cooling capacity with compensatory actions taken per existing procedures during elevated ambient conditions. There is no increase in heat load to the CRCWS. Therefore, the CW system, including the DCWS and CRCWS, is adequate to support operation at TPO uprate conditions.

6.4.4 Turbine Building Closed Cooling Water System

The power-dependent heat loads on the TBCCW system increased by the TPO are those related to the operation of the iso-phase bus duct cooler. The remaining TBCCW heat loads are not dependent upon reactor power and do not increase. The TBCCW system has been evaluated at 102% of CLTP and has sufficient capacity to ensure that adequate heat removal capability is available for TPO operation. Therefore, TPO uprate has no effect on the design of the TBCCW system.

6.4.5 Reactor Building Closed Cooling Water System

The heat loads on the RBCCW system do not increase significantly due to the TPO uprate. The most significant RBCCW heat loads during normal operation are those related to the operation of the RWCU non-regenerative heat exchangers and the reactor recirculation pumps. These heat loads do not increase with TPO uprate. The RBCCW system experiences a slight heat load increase associated with backup fuel pool cooling during refueling activities; however, the system has adequate design margin to remove the additional heat evaluated to a bounding 102% of CLTP. Therefore, the RBCCW system is acceptable for the TPO uprate.

6.4.6 Emergency Heat Sink

A review was performed to evaluate the increased emergency heat sink (EHS) heat load for the TPO. The ECT contains sufficient inventory of water to achieve and maintain safe shutdown of both units' reactors for seven days without makeup. Based on the water consumed during continuous cooling tower operation at the rated flow conditions for seven days, the ECT reservoir margin is reduced from ~17.3% at CLTP to ~15.7% for TPO. The current TS for the EHS limits are adequate due to conservatism in the original design.

6.5 STANDBY LIQUID CONTROL SYSTEM

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that all or some of the control rods cannot be inserted. This manually operated system pumps a highly enriched sodium pentaborate solution into the vessel to achieve a subcritical condition. A plant-specific evaluation concludes that the TPO uprate does not affect shutdown or injection capability of the SLCS. Because the shutdown margin is reload dependent, the shutdown margin and the required reactor boron concentration are confirmed for each reload core.

The ATWS evaluation in Section 9.3.1 shows that the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1,207 psia during the time the SLCS is analyzed to be in operation. This evaluation shows the pressure margin for the SLCS pump discharge relief valves is 183 psi, which is adequate to ensure that the SLCS relief valves remain closed during system injection. The minimum reactor pressure, just prior to the time when SLCS initiates, remains low enough to ensure SLCS relief valve closure prior to the analyzed SLCS initiation time in the event of an early initiation of the SLCS during the initial ATWS transient pressure response. Therefore, SLCS operation during an ATWS at the TPO power level is acceptable considering the MELLLA+ operating domain expansion.

The evaluation shows that the TPO uprate has no adverse effect on the ability of the SLCS to mitigate an ATWS.

6.6 POWER-DEPENDENT HEATING, VENTILATION AND AIR CONDITIONING

The heating, ventilation and air conditioning (HVAC) systems that are potentially affected by the TPO uprate consist mainly of heating, cooling supply, exhaust, and recirculation units in the Turbine Building, Reactor Building, steam tunnel and primary containment (drywell).

TPO results in a minor increase in the MS tunnel heat load caused by the slightly higher FW process temperature (1°F to 2°F increase). The increased heat load results in an insignificant (~0.03°F) increase in MS tunnel area temperature with TPO and will remain within the system cooling capacity. Outside of the MS tunnel, heat loads in the Reactor Building will not experience any change with the TPO uprate. In the drywell, the increase in heat loads and area temperature are insignificant (< 0.1°F increase) with TPO and will remain within the system cooling capacity with compensatory actions taken per existing procedures during elevated ambient conditions. In the Turbine Building, the temperature increases are very low (maximum of ~0.4°F increase) due to the increase in the FW and BOP process temperatures. Other areas are unaffected by the TPO uprate because the process temperatures and electrical heat loads remain constant.

Therefore, the power-dependent HVAC systems are adequate to support the TPO uprate.

6.7 FIRE PROTECTION

Operation of the plant at the TPO RTP level does not affect the fire suppression or detection systems. There is no change in the physical plant configuration or combustible loading resulting from the TPO uprate.

The operator manual actions that are being used for compliance with 10 CFR 50, Appendix R were reviewed. No new operator actions have been identified in areas where environmental conditions, such as heat, would challenge the operator or would become a challenge with TPO conditions. Because this uprate is being performed at a constant pressure and temperature, the normal temperature environments are not affected by TPO. Therefore, the operator manual actions required to mitigate the consequences of a fire are not affected.

A review was conducted of the Fire Protection Program as related to administrative controls, fire barriers, fire protection responsibilities of plant personnel and resources necessary for systems required to achieve and maintain safe-shutdown. The review looked at the effect of TPO uprate and how it would affect these areas. The TPO uprate will have no effect on fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel and resources necessary for systems required to achieve and maintain safe-shutdown.

A review was conducted of all repair activities that are credited to obtain and maintain cold shutdown. The PBAPS Appendix R analysis demonstrates that the station can reach cold shutdown with significant margin to the 72-hour requirements in 10 CFR 50 Appendix R, Sections III.G.1.b and III.L. The TPO and the additional decay heat removal would not affect the ability to reach and maintain cold shutdown within 72 hours.

PBAPS does not take credit in any safety analysis for the fire protection system other than for fire protection activities. Procedures are provided under Transient Response Implementation Plan procedures, Severe Accident Management Procedures, Extensive Damage Mitigation Guidelines, and FLEX Support Guidelines, which provide instructions for utilizing fire protection system pumps to provide water to the reactor, the drywell, the spent fuel pool, or the suppression chamber if necessary. However, this use of the non-safety-related fire protection system is not credited in any safety analyses and TPO uprate operation will not require any changes to these procedures regarding the utilization of the fire protection system.

Therefore, the fire protection systems and analyses are not affected by the TPO uprate.

6.7.1 10 CFR 50 Appendix R Fire Event

The 10 CFR 50 Appendix R fire safe shutdown (FSSD) events were previously analyzed in the PUSAR (Reference 14), Section 2.5.1.4.2. A plant-specific analysis was performed for PBAPS at TPO RTP conditions to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R. The fuel heatup analysis was performed using the SAFER/PRIME-LOCA analysis model. The containment analysis was performed using the SHEX model.

Two limiting shutdown methods, A and C, defined in the PBAPS PUSAR (Reference 14) were reanalyzed under TPO conditions. The bounding PCT for PBAPS is shutdown Method C with one RHR in LPCI mode. The bounding peak suppression pool temperature for PBAPS is shutdown Method A with RCIC, and one RHR in LPCI mode.

The results of the Appendix R evaluation at TPO conditions in Table 6-8 demonstrate that the fuel cladding integrity, reactor vessel integrity, and containment integrity are maintained. One train of systems remains available to achieve and maintain safe shutdown conditions. For FSSD Method C1, the time from event initiation to the point at which conditions require manual initiation of RPV depressurization from the control room is slightly reduced, but is more than the time available for other FSSD methods requiring manual RPV depressurization. The time line for manual operator actions, including FSSD Method C1, and the associated approved 10 CFR 50 Appendix R exemptions required for the FSSD analysis are unaffected. There are no required changes to the operating procedures or implementing strategy for the TPO uprate.

Therefore, the PBAPS analysis results for 10 CFR 50 Appendix R FSSD events at TPO RTP conditions are acceptable.

6.8 SYSTEMS NOT AFFECTED BY TPO UPRATE

Based on experience and previous NRC reviews, all systems that are significantly affected by TPO are addressed in this report. Systems not addressed by this report are not significantly affected by TPO. The systems unaffected by TPO at PBAPS are confirmed to be consistent with the descriptions provided in the TLTR.

Table 6-1 Plant Electrical Equipment Ratings

Parameter	Value
Generator Ratings	
Generator Output (MWe)	1,408 (Unit 2) 1,377 (Unit 3)
Rated Voltage (kV)	22 (both units)
Power Factor	0.920 (Unit 2) 0.900 (Unit 3)
Generator Output (MVA)	1,530 (both units)
Current Output (Amps)	40,152 (both units)
Isolated Phase Bus Duct Rating (Amps)	
Generator Bus	21,200
Main Section	42,300
Delta Section	20,500
Auxiliary Section	2,000
Main Transformers Rating (MVA)	1,530

Table 6-2 Main Generator Ratings Comparison

Power Level	Design	Maximum Nominal			
		Unit 2		Unit 3	
	MVA @ 75 psig H_2	MWe @ 75 psig H_2	MVAR @ 75 psig H_2	MWe @ 75 psig H_2	MVAR @ 75 psig H_2
CLTP	1,530	1,408	600	1,377	667
TPO RTP (1)	1,530	1,408	600	1,377	667

Note:

(1) Operation at the TPO uprated condition is not expected to have any adverse effect on the operation of the main generators. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures. Existing PBAPS operating procedures are in place to ensure the generator's design rating of 1,530 MVA is not exceeded.

Table 6-3 Main Transformer Ratings Comparison

Power Level	Design MVA at 65°C	MVA Loading
CLTP	1,530	1,530
TPO RTP (1)	1,530	1,530

Note:

(1) Operation at the TPO uprated condition is not expected to have any effect on the operation of the main transformer.

Table 6-4 Unit Auxiliary Transformer Ratings Comparison

Component	Component Rating (MVA)	CLTP Duty (MVA)	CLTP Margin (%)	TPO Duty (MVA)	TPO Margin (%)
Unit 2 Auxiliary Transformers	45.4	43.90	3.3	43.91	3.3
Unit 3 Auxiliary Transformers	45.4	41.52	8.6	41.53	8.5

Table 6-5 Start-Up and Emergency Auxiliary Power Transformer Ratings Comparison

Component	Component Rating (MVA)	CLTP Duty (MVA)	CLTP Margin (%)	TPO Duty (MVA)	TPO Margin (%)
Start-Up and Emergency Auxiliary Transformers (Maximum MVA)	50.0	48.3	3.4	48.3	3.4

Note:

(1) Operation at the uprated condition is not expected to have any effect on the operation of the start-up and emergency auxiliary power transformers. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

Table 6-6 FPCCS Parameters

Parameter	CLTP	TPO
Number of RHR/FPCCS trains	1 / 3	1 / 3
RHR heat exchanger flow rate, RHR/HPSW	5,000 / 4,500 gpm	5,000 / 4,500 gpm
Fuel pool heat exchanger flow rate, SFP/SW	555 / 800 gpm	555 / 800 gpm
Design RHR heat removal capability (per heat exchanger)	43.9E+6 BTU/hr	43.9E+6 BTU/hr
Design FPCCS heat load (per heat exchanger)	3.75E+6 BTU/hr	3.75E+6 BTU/hr
Fuel cycle (months)	24	24
Bulk pool temperature (during refueling)	< 150°F	< 150°F

Table 6-7 FPCCS Response at CLTP and CLTP x 1.02

Normal Offload, Full Cooling Capability								
3 FPCCS pumps, 3 FPCCS Heat Exchangers (HXs). 1,665 gpm total SFP flow, 2,400 gpm total SW flow.								
SW Temp (°F)	**Start of Offload (hours after shutdown)**		**Maximum SFP Temperature (°F)**		**Time to Boil from Maximum Temperature (hours)**		**Makeup Flow Required at Boiling (gpm)**	
	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02
90	90	100	140	140	8.1	8.1	48	48
Full-Core Offload, RHR Fuel Pool Cooling Assist Mode								
1 RHR pump, 1 RHR HX. 5,000 gpm SFP flow, 4,500 gpm HPSW flow.								
HPSW Temp (°F)	**Start of Offload (hours after shutdown)**		**Maximum SFP Temperature (°F)**		**Time to Boil from Maximum Temperature (hours)**		**Makeup Flow Required at Boiling (gpm)**	
	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02
92	150	150	140	141	4.2	4.2	88	90
Normal Offload, Single Failure								
2 FPCCS pumps, 2 FPCCS HXs. 1,110 gpm total SFP flow, 1,600 gpm total SW flow.								
SW Temp (°F)	**Start of Offload (hours after shutdown)**		**Maximum SFP Temperature (°F)**		**Time to Boil from Maximum Temperature (hours)**		**Makeup Flow Required at Boiling (gpm)**	
	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02	CLTP[1]	CLTP X 1.02
90	210	230	150	150	8.7	8.7	40	40

Note:

(1) The FPCCS evaluation has been re-performed using a corrected SFP volume. See Section 6.3.1.

Table 6-8 Appendix R Fire Event Evaluation Results

Parameter	TPO 1 LPCI, 3 ADS (Shutdown Method C) [1]	TPO RCIC, 1 LPCI (Shutdown Method A) [1]	Appendix R Criteria
Peak Fuel Cladding Temperature (°F)	1,485	No core heat-up [2]	< 1,500°F
Maximum Operator Action Time to Open ADS valves (minutes)	24.9	Not Calculated [3]	See Note 4
Peak RPV Dome Pressure (psig)	1,145.3	1,145.3	≤ 1,325
Peak Suppression Pool Bulk Temperature (°F)	204	205	< 281

Notes:

(1) SAFER/PRIME-LOCA and SHEX methodologies used.

(2) Initial steady-state fuel cladding temperature.

(3) Controlled depressurization with rate of 100°F/ hour starting at 210 minutes after event initiation.

(4) The maximum ADS actuation time should allow the core to remain covered with a short fuel uncovery period permitted, providing the PCT acceptance criterion is met.

7.0 POWER CONVERSION SYSTEMS

7.1 TURBINE-GENERATOR

The PBAPS main turbine is designed with a maximum (VWO) steam flow capacity in excess of TPO rated conditions to ensure that the design rated output is achieved. The excess capacity ensures that the main turbine can meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may affect the flow-passing capability of the unit.

Refer to Section 5.2.1 for additional discussion of effective throttle flow margin and reactor pressure control.

For the TPO uprate condition 4,016 MWt (approximately 101.66% of CLTP), the rated throttle steam flow is increased to 16.467 Mlb/hr at an approximate throttle pressure of 955 psia. The evaluated increased throttle steam flow is 101.8% of current rated steam flow. The evaluated increased throttle flow is due to the steam flow increase associated with operation at TPO uprate conditions.

At TPO RTP, the main generator is projected to produce an electrical power output of approximately 1,387.9 MWe. The main generator will be operated within its design capability at TPO uprate conditions. Main generator ratings for electrical and reactor power are discussed in Section 6.1.1.

Heat balances were prepared to determine the TPO uprate turbine steam path conditions. The turbine and generator stationary and rotating components are evaluated at TPO uprate conditions and found to be acceptable. The increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations resulting from operation at TPO uprate conditions are within the DLs of the T/G systems and components; therefore, they are acceptable at the TPO uprate condition. The results of these evaluations show that no additional physical modifications are needed to support operation at the TPO uprate condition.

The existing rotor missile analysis uses fracture mechanics, stress analysis, and probability theory to determine the risk of rotor failure to the plant following NRC guidelines. The original analysis found the risk of low pressure rotor body failure was only about one tenth of that permitted by NRC criteria. Operation at TPO RTP essentially has no effect on this conclusion and is therefore acceptable. The high pressure turbine is an integral forged rotor design which requires no further turbine missile analysis.

The overspeed evaluation determines the peak overspeed that the rotor train would be expected to reach following a load rejection. This analysis uses VWO conditions to bound the available stored energy in the cycle associated with TPO uprate conditions and is not affected by the TPO uprate.

7.2 CONDENSER AND STEAM JET AIR EJECTORS

The main condenser capability was evaluated for performance at the TPO uprate conditions in Section 5.3.15. Air leakage into the condenser does not increase as a result of the TPO uprate.

The small increase in hydrogen and oxygen flows from the reactor core does not affect the steam jet air ejectors (SJAEs) because the design was based on flows greater than required flows at TPO uprate conditions. Therefore, the condenser air removal system is not affected by the TPO uprate and the SJAEs are adequate for operation at the TPO conditions.

7.3 TURBINE STEAM BYPASS

The turbine steam bypass valves operate at a steam flow capacity of approximately 22.39% of the 100% rated flow at CLTP. The steam bypass capacity at the TPO RTP is approximately 21.96% of the 100% TPO RTP steam flow rate. The steam bypass system is non-safety-related. While the bypass capacity as a percent of rated steam flow is reduced, the actual steam bypass capacity is unchanged.

7.4 FEEDWATER AND CONDENSATE SYSTEMS

The condensate and FW systems are designed to provide FW at the temperature, pressure, quality, and flow rate required by the reactor. These systems are not safety-related outside of the FW containment isolation boundary; however, their performance may affect the plant availability and capability to operate reliably at the TPO uprate condition.

A review of the PBAPS FW heaters, heater drain system, condensate demineralizers, and the pumps (condensate and FW) demonstrated that the components are capable of providing the slightly higher TPO uprate FW flow rate at the desired temperature and pressure. A review of the PBAPS heater drain system demonstrated that the components are capable of supporting the slightly higher TPO uprate extraction flow rates.

Performance evaluations were based on an assessment of the capability of the condensate and FW systems and equipment to remain within the design limitations of the following parameters:

- Ability to avoid suction pressure trip
- Flow capacity
- Rated motor horsepower.

7.4.1 Normal Operation

The reactor feedwater pumps (RFPs) will provide FW at the required flow rate and with sufficient RPV interface pressure to support the TPO uprate. This is accomplished by slightly increasing the RFP speed to increase the FW flow rate while still providing sufficient pressure at the RPV interface. During steady-state conditions, the condensate and FW systems have available NPSH for all of the pumps to operate without cavitation at the TPO uprate conditions. Adequate margin during steady-state conditions also exists between the calculated minimum pump suction pressure and the low suction pressure trip setpoints.

The existing FW design pressure and temperature requirements bound operating conditions with adequate margin. The FW heaters are ASME Section VIII pressure vessels. The heaters were verified to be acceptable for the slightly higher FW heater temperatures and pressures for the TPO uprate.

7.4.2 Transient Operation

To account for FW demand transients, the condensate and FW systems were evaluated to ensure that sufficient margin above the TPO uprated flow is available. For system operation with all system pumps available, the predicted operating parameters were acceptable and within the component capabilities.

The condensate and FW systems provide adequate flow delivery following a single RFP trip without causing a reactor shutdown. Operation at the TPO condition continues to support this capability.

7.4.3 Condensate Demineralizers

The effect of the TPO uprate on the condensate filter/demineralizer (CFD) system was reviewed. The system can accommodate (without bypass) TPO uprate conditions while operating with one CFD vessel removed from service (when backwash/resin change out is required).

The effect of the TPO uprate on the CFDs was reviewed. The flow rate through the condensate system increases by up to 2.2% from the current rated flow, but remains within the design flow rate. The CFDs experience slightly higher loadings at the TPO RTP level, which results in slightly reduced run times. However, the reduced run times are acceptable because a spare unit is utilized when cleaning is required (refer to Section 8.0 for the effect on the radwaste system). Reduced run times (more frequent cleaning) of polisher units does not affect CFD system capacity.

8.0 RADWASTE AND RADIATION SOURCES

8.1 LIQUID AND SOLID WASTE MANAGEMENT

The liquid radwaste system collects, monitors, processes, stores, and returns processed liquid radioactive waste to the plant for reuse, discharge, or shipment.

The solid radwaste system collects, monitors, processes, and stores processed solid radioactive waste prior to offsite disposal.

Major sources of liquid and wet solid waste are from the CFDs. The TPO uprate results in an approximately 2% increase in flow rate through the condensate system. This potentially results in a reduction in the average time between backwashes of the condensate pre-filters and replacement of the condensate demineralizer resin. This potential reduction of condensate demineralizer service time does not affect plant safety.

The floor drain collector subsystem and the waste collector subsystem both receive periodic inputs from a variety of sources. Neither subsystem experiences a significant increase in volume due to operation at the TPO uprate condition.

The total volume of processed waste is not expected to increase appreciably. The only significant increase in processed waste is due to the more frequent backwashes of the CFDs. A review of plant operating effluent reports and the slight increase expected from the TPO uprate leads to the conclusion that the requirements of 10 CFR 20 and 10 CFR 50, Appendix I will continue to be met. Therefore, the TPO uprate does not adversely affect the processing of liquid or solid radwaste, and there are no significant environmental effects.

8.2 GASEOUS WASTE MANAGEMENT

The gaseous waste management systems collect, control, process, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system and various building ventilation systems. The systems are designed to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

Non-condensable radioactive gas from the main condenser normally contains activation gases and fission product radioactive noble gas parents. These are the major sources of radioactive gas and are greater than all other sources combined. These non-condensable gases, along with non-radioactive air in-leakage, are continuously removed from the main condensers by the SJAEs that discharge into the offgas system.

Building ventilation systems control airborne radioactive gases by using components such as high efficiency particulate air (HEPA) and charcoal filters, and radiation monitors that activate isolation dampers or trip supply and exhaust fans, or by maintaining negative or positive air pressure to limit migration of gases. The changes to the gaseous radwaste releases are proportional to the change in core power, and the total releases are a small fraction of the design basis releases.

The release limit is an administratively controlled variable and is not a function of core power. The gaseous effluents are well within limits at CLTP operation and remain well within limits following implementation of the TPO uprate; therefore, there are no significant environmental effects from gaseous effluents due to the TPO uprate.

The offgas system was evaluated for the TPO uprate. Radiolysis of water in the core region, which forms H_2 and O_2, increases linearly with core power, thus increasing the volume of waste gas processed by the recombiner and related components. The original design basis radiolytic gas production rate at OLTP yields a H_2 flow rate of 149.3 cfm (with a corresponding stoichiometric O_2 contingent of 74.6 cfm). The proportional H_2 flow rate for the TPO uprate thermal power is 139.3/123.2 cfm (Unit 2/Unit 3). The corresponding stoichiometric O_2 flow rates are 69.6/61.6 cfm (Unit 2/Unit 3). The increase in H_2 and O_2 due to the TPO uprate remains well within the capacity of the system. Therefore, the TPO uprate does not adversely affect the offgas system design or operation

8.3 RADIATION SOURCES IN THE REACTOR CORE

TLTR Appendix H describes the methodology and assumptions for the evaluation of radiological effects for the TPO uprate.

During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy released per unit of reactor power. Therefore, for TPO, the percent increase in the operating source terms is no greater than the percent increase in power. The PBAPS-specific source term increases due to the TPO uprate are bounded by the safety margins of the design basis sources.

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term is defined in terms of million electron volts (MeV)/sec per watt of reactor thermal power (or equivalent) at various times after shutdown. Therefore, the total gamma energy source increases in proportion to reactor power.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These are needed for post-accident and SFP evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops “equilibrium” activities in the fuel (typically three years). Most radiologically significant fission products reach equilibrium within a 60-day period. The calculated inventories are approximately proportional to core thermal power. Consequently, for TPO, the inventories of those radionuclides, which reached or approached equilibrium, are expected to increase in proportion to the thermal power increase. The inventories of the very long-lived radionuclides, which did not approach equilibrium, are both power and exposure dependent. Thus, the long-lived radionuclides are expected to increase proportionally to power. The radionuclide inventories are provided in terms of curies per megawatt of reactor thermal power at various times after shutdown.

8.4 RADIATION SOURCES IN REACTOR COOLANT

8.4.1 Coolant Activation Products

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation is the dominant source in the Turbine Building and in the lower regions of the drywell. Because these sources are produced by interactions in the core region, their rates of production are proportional to power. However, the concentration in the steam remains nearly constant, because the increase in activation production is balanced by the increase in steam flow. As a result, the activation products, observed in the reactor water and steam, increase in approximate proportion to the increase in thermal power.

8.4.2 Activated Corrosion Products

The reactor coolant contains activated corrosion products from metallic materials entering the water and being activated in the reactor region. Under the TPO uprate conditions, the activation rate in the reactor region increases with power, and the filter efficiency of the condensate demineralizers may decrease. The net result may be an increase in the activated corrosion product production. However, total TPO activated corrosion product activity levels in the reactor water remain less than the design basis activated corrosion product activity. Therefore, no change is required in the design basis activated corrosion product concentrations for the TPO uprate.

8.4.3 Fission Products

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. The noble gases released during plant operation result from the escape of minute fractions of the fission products from the fuel rods. Noble gas release rates increase approximately with power level. This activity is the noble gas offgas that is included in the PBAPS design. The total offgas rates for TPO uprate operations are bounded by the CLTP analysis, which was performed at 102% CLTP.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. As is the case for the noble gases, there is no expectation that releases from the fuel increase due to the TPO uprate. Activity levels in the reactor water at TPO conditions are approximately equal to current measured data, which are fractions of the design basis values. Therefore, the design basis values are unchanged.

8.5 RADIATION LEVELS

Normal operation radiation levels increase slightly for the TPO uprate. PBAPS was designed with sufficient margin for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the design, source terms, and analytical techniques.

Radiation levels in most areas of the plant increase by no more than the percentage increase in power level. In a few areas near the reactor water piping where accumulation of corrosion product crud is expected, as well as near some liquid radwaste equipment, the increase could be slightly higher.

Regardless, individual worker exposures will be maintained within acceptable limits by the site ALARA program, which controls access to radiation areas. The PBAPS radiation protection program procedural controls compensate for any minor increase in radiation levels due to the TPO uprate.

The change in core activity inventory resulting from the TPO uprate (Section 8.3) increases post-accident radiation levels by no more than approximately the percentage increase in power level. Previous analyses of post-accident radiation levels were performed at 102% of CLTP and therefore bound the effects of the TPO uprate on the plant and the habitability of the on-site emergency response facilities. A review of areas requiring post-accident occupancy concluded that access needed for accident mitigation is not significantly affected by the TPO uprate.

Section 9.2 addresses the main control room doses for the worst-case accident.

8.6 NORMAL OPERATION OFF-SITE DOSES

A review of the normal radiological effluent doses shows that at CLTP the public dose effects of design basis gaseous and liquid releases remain within 10 CFR 50, Appendix I limits with substantial margin. The TPO uprate does not involve significant increases in the offsite dose from noble gases, airborne particulates, iodine, tritium or liquid effluents. In addition, radiation from shine is not a significant exposure pathway. Present offsite radiation levels are a negligible portion of background radiation. Therefore, the normal offsite doses are not significantly affected by operation at the TPO RTP level and remain below the limits of 10 CFR 20 and 10 CFR 50, Appendix I.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 ANTICIPATED OPERATIONAL OCCURRENCES

[[

]] The standard reload analyses consider the plant conditions for each fuel cycle.

9.1.1 Alternate Shutdown Cooling Evaluation

The capability of the alternate shutdown cooling method to achieve cold shutdown within 36 hours was analyzed at 102% of CLTP and ANS/ANSI 5.1-1979 with 2-sigma adders decay heat (Reference 14). This bounds the TPO power level and therefore there is no change to the current PBAPS licensing basis.

9.2 DESIGN BASIS ACCIDENTS

The radiological consequences of a DBA are proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanisms from the core to the release point. The radiological releases at the TPO uprate power are generally expected to increase in proportion to the core inventory increase, which is in proportion to the power increase.

Postulated DBA events have been evaluated and analyzed to show that NRC regulations are met for 2% above the CLTP. DBA events have either been previously analyzed at 102% of CLTP, which bounds the TPO power level, or are not dependent on core thermal power. The MSLBA outside containment was evaluated using a 4 μCi/g dose equivalent I-131 limit on reactor coolant activity. The limit on reactor coolant activity is unchanged for the TPO uprate condition. The evaluation/analysis was based on the methodology, assumptions, and analytical techniques described in the PBAPS current licensing basis, regulatory guides, and in previous SEs.

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transient Without Scram

Plant-specific EPU (Reference 14) and MELLLA+ (Reference 15) analyses were previously done at CLTP. An additional plant-specific analysis of the limiting ATWS events, consistent with the M+SAR (Reference 15) was performed at the TPO bounding high thermal power of 4,018 MWt. The assumptions and approaches in the analysis are consistent with those stated in Reference 15.

The plant-specific ATWS analysis confirmed that the following ATWS acceptance criteria are met:

- Maintain reactor vessel integrity (i.e., peak vessel bottom pressure less than the ASME Service Level C limit of 1,500 psig).
- Maintain containment integrity (i.e., maximum containment pressure and temperature less than the limiting pressure (56 psig) and temperature (180°F) of the containment structure).
- Maintain coolable core geometry.

The TPO RTP ATWS analysis was performed using the NRC-approved codes PANAC, ODYN, TASC, and TRACG (see Table 1-1a). The key inputs to the ATWS analysis are provided in Table 9-1. The results of the analysis are provided in Table 9-2.

M+LTR SER Limitation and Condition 12.23.2 requires that the plant-specific automatic settings be modeled for ATWS. For PBAPS, the plant automatic settings, which include the ATWS-RPT, low pressure isolation, and SRV actuation, are modeled based on the input parameters in Table 9-1.

As required by M+LTR SER Limitation and Condition 12.23.8, the plant-specific ATWS analyses account for plant- and fuel-design-specific features. The ATWS analyses are performed based on GNF2 fuel designs from PBAPS Unit 2. This core is representative for addressing any cores of GNF2 fuel for both PBAPS Units 2 and 3.

In accordance with M+LTR SER Limitation and Condition 12.23.3, the plant-specific ATWS analyses assumed SRV setpoints that account for a 3% drift tolerance. PBAPS as-found SRV lift setpoint tests do not show a propensity for setpoint drift higher than the 3% drift tolerance. Therefore, the SRV upper tolerances used in the ATWS analyses are consistent with the plant-specific performance.

The ATWS overpressure and ATWS with core instability events for PBAPS MELLLA+ were evaluated using a plant-specific TRACG model. In accordance with Methods LTR SER Limitation and Condition 9.20, the void reactivity coefficients bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core.

The results of the ATWS analysis meet the ATWS acceptance criteria. Therefore, the PBAPS response to an ATWS event at TPO RTP is acceptable. Thermal-hydraulic instability in conjunction with ATWS events is evaluated in Section 9.3.1.4 and meets the acceptance criteria.

PBAPS also meets the ATWS mitigation requirements defined in 10 CFR 50.62:

- Installation of an alternate rod insertion system;
- Boron injection equivalent to 86 gpm; and
- Installation of automatic RPT logic (i.e., ATWS-RPT).

There are no changes to the assumed operator actions or response times for the TPO RTP ATWS analysis.

When required by changes in plant configuration (as identified by the design change process), changes to emergency operating procedures (EOPs), including changes to EOP calculations and plant data, are developed and implemented in accordance with the plant administrative procedures for EOP program maintenance.

TPO implementation does not significantly change the transient sequence of events. Therefore, there is no change in operator strategy on ATWS level reduction or early boron injection. TPO may affect some of the calculated curves, but does not affect stability mitigation actions.

9.3.1.1 ATWS (Overpressure) - TRACG

The higher operating steam flow results in slightly higher peak vessel pressures.

The overpressure evaluation includes consideration of the most limiting RPV overpressure case. Four ATWS events: (1) MSIVC; (2) pressure regulator failure open (PRFO); (3) inadvertent opening of a relief valve (IORV); and (4) loss of offsite power (LOOP) are considered.

TRACG ATWS Overpressure LTR SER Limitation and Condition 4.2 requires reporting the plant-specific power-to-flow ratio at rated power and minimum CF (Reference 50). Additionally, TRACG ATWS Overpressure LTR SER Limitation and Condition 4.3 mandates the actual power level be stated from the TRACG ATWS application. [[

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The MSIVC and PRFO cases were performed for PBAPS. The IORV cases and LOOP cases are non-limiting.

The limiting pressure results are given in Table 9-2, and the analysis results for the PRFO and MSIVC events are shown in Figures 9-1 and 9-2, respectively. The ATWS (Overpressure) results meet the vessel pressure acceptance criterion.

9.3.1.2 ATWS (Suppression Pool Pressure and Temperature) - ODYN

The higher power and decay heat results in slightly higher suppression pool pressures and temperatures. Consistent with M+LTR SER Limitation and Condition 12.23.10, the PBAPS plant-specific ATWS analysis contains information relevant to any increase in containment pressure during the event.

The suppression pool pressure and temperature evaluation includes consideration of the most limiting RHR pool cooling capability case. Four ATWS events: (1) MSIVC; (2) PRFO; (3) IORV; and (4) LOOP are considered.

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M+LTR SER Limitation and Condition 12.23.11 requires that the use of suppression pool temperature limits higher than the heat capacity temperature limit (HCTL) for emergency depressurization must be justified. The containment DL is the ATWS acceptance criteria. [[

]] A best-estimate TRACG analysis modeling emergency depressurization is not required if the suppression pool temperature from the licensing basis ODYN long-term calculation remains below the HCTL.

The MSIVC and PRFO cases were performed for PBAPS. The IORV cases and LOOP cases are non-limiting. The key inputs to the ATWS analysis are provided in Table 9-1. The limiting analysis results are given in Table 9-2. The MSIVC and PRFO sequence of events are given in Tables 9-3 and 9-4, respectively. The analysis results for the PRFO and MSIVC events are shown in Figures 9-3 through 9-10. The ATWS (suppression pool pressure and temperature) limiting events meet all acceptance criteria.

9.3.1.3 ATWS (Peak Cladding Temperature) – ODYN/TASC

The limiting PCT is given in Table 9-2. Note that the PCT at TPO is less than the PCT at CLTP due to more favorable CF conditions at TPO.

For ATWS events, the acceptance criteria for PCT and local cladding oxidation for emergency core cooling systems, defined in 10 CFR 50.46, are adopted to ensure an ATWS event does not impede core cooling.

For TPO, the fuel PCT during an ATWS event is 1,483ºF, local cladding oxidation is < 17%, and coolable geometry is ensured. Therefore, ATWS PCT is in compliance with the acceptance criteria of 10 CFR 50.46; subsequently, coolable core geometry is ensured by meeting the 2,200ºF PCT and the 17% local cladding oxidation acceptance criteria stated in 10 CFR 50.46.

9.3.1.4 ATWS with Core Instability - TRACG

The ATWS with core instability event current analysis-of-record is presented in Section 9.3.3 of the M+SAR (Reference 15), as supplemented by request for additional information (RAI) SRXB-RAI-18 response provided in Reference 51. The same analysis was performed for TPO RTP, and the conclusions remain the same; a coolable geometry is maintained and the ATWS acceptance criteria remains satisfied. This result is an expected outcome because both the MELLLA+ and TPO ATWSI events initiate from the same rod line and therefore reduce to nearly identical power, flow, and pressure conditions following the RPT and prior to instabilities resulting in a small change in PCT (10°F) that is considered insignificant.

The initial power, even power that exceeds 120%, and CF are not directly important for an ATWSI event. The important parameters are the power, flow, and pressure conditions after a recirculation pump trip (in both turbine trip with bypass (TTWBP) and RPT ATWSI events). Because the MELLLA+ and TPO initiate from the same rod line (MELLLA+ boundary), the thermal-hydraulic conditions after the RPT will be approximately the same; therefore, the severity will also be approximately the same.

M+LTR SER Limitation and Condition 12.19 requires that a plant-specific ATWS instability calculation be performed to demonstrate that PBAPS EOP actions, including boron injection and water level control strategy, effectively mitigate an ATWS event with large power oscillations in the MELLLA+ operating domain. This plant-specific analysis was performed for MELLLA+ in Reference 15. A plant-specific ATWSI analysis at TPO RTP was also performed. This analysis was: (1) based on the limiting of BOC, peak reactivity exposure condition (MOC), and EOC; (2) modeled the plant-specific configuration important to the ATWSI response; and (3) used the limiting of the regional mode or core-wide mode nodalization scheme. M+LTR SER Limitation and Condition 12.23.5 requires that the power density be less than 52.5 MWt/Mlbm/hr. For PBAPS, the plant-specific maximum power-to-flow ratio at rated power and minimum CF is 46.0 MWt/Mlbm/hr. This value for the maximum power-to-flow ratio meets the requirement and is less than the Grand Gulf Nuclear Station MELLLA+ power-to-flow ratio (49.0 MWt/Mlbm/hr) (Section 9.3.3 of Reference 52).

The plant-specific TRACG calculation modeled in-channel water rod flow in accordance with M+LTR SER Limitation and Condition 12.24.1. The plant-specific ATWSI calculation was performed using the latest NRC-approved neutronic and thermal-hydraulic codes TGBLA06/PANAC11 and TRACG04 (Reference 13). A GNF2 equilibrium core was used for the calculation and this complies with M+LTR SER Limitation and Condition 12.3.d. The TRACG ATWSI analysis results are included in this section in compliance with M+LTR SER Limitation and Condition 12.23.6.

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The limiting PCT results of the plant-specific TRACG ATWSI calculation are provided in Table 9-2. Figures 9-11 through 9-14 show the mitigating effect of decreasing water level for the TTWBP and RPT ATWSI events.

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The results of the plant-specific TRACG ATWSI calculation meet the ATWS review criteria. Therefore, the PBAPS response to an ATWSI event initiated at TPO RTP and MELLLA+ conditions is acceptable. PBAPS EOP actions, including boron injection and water level control strategy, effectively mitigate an ATWS event with large power oscillations in the MELLLA+ operating domain.

Therefore, the PBAPS analysis results at TPO RTP conditions are acceptable.

9.3.1.5 SLCS Performance and Hardware

The increased core power and reactor steam flow rates, in conjunction with the SRV capacity and response times, could affect the capability of the SLCS to mitigate the consequences of an ATWS event. Based on the results of the plant-specific ATWS analysis, the maximum reactor lower plenum pressure following the limiting ATWS event reaches 1,207 psia during the time the SLCS is analyzed to be in operation. Compared to the results documented in the M+SAR (Reference 15; 1,206 psia), there is an insignificant difference in lower plenum pressure. Consequently, the pump discharge pressure and operating pressure margin for the pump discharge relief valves remain about the same. This conclusion complies with M+LTR SER Limitation and Condition 12.23.9.

The SLCS ATWS performance is evaluated for a representative core design for TPO. The evaluation shows that TPO has no adverse effect on the ability of the SLCS to mitigate an ATWS. There are no changes in operator action or response timing for TPO for PBAPS, and the

ATWS analysis confirms acceptable results. Therefore, the system performance and hardware meet all CLTR (Reference 6) dispositions.

9.3.1.6 Suppression Pool Temperature following ATWS Event

The boron injection rate requirement for maintaining the peak suppression pool water temperature limits, following the limiting ATWS event with SLCS injection, is not increased for TPO. Therefore, the suppression pool temperature following an ATWS event meets all CLTR (Reference 6) dispositions.

9.3.1.7 Equipment Out-of-Service and Flexibility Options

The following flexibility options and/or equipment OOS options are considered in the evaluation:

- MELLLA+ (83% at CLTP or 85.2% at TPO RTP)
- ICF (110%)
- FFWTR (90°F Reduction) – not allowed in the MELLLA+ domain
- FWHOOS (55°F Reduction) – not allowed in the MELLLA+ domain
- FWHOOS (10°F Reduction) – allowed in the MELLLA+ domain per Operating License Condition 2.C(16)
- SLO (2,701 MWt; Core Flow of 57.4 Mlbm/hr) – not allowed in the MELLLA+ domain
- TBV OOS
- RPT OOS
- 1 SRV/Spring Safety Valve (SSV) OOS
- TSV/TCV OOS
- MSIV OOS (≤ 75% of 3,514 MWt)
- PR OOS
- PLU OOS
- ARTS Program
- 24 Month Cycle.

9.3.2 Station Blackout

The SBO event was previously analyzed in Section 2.3.5 of the PUSAR (Reference 14). A plant-specific analysis was performed for PBAPS confirming continued compliance to 10 CFR 50.63 at TPO RTP conditions.

The following major characteristics that affect the ability to cope with an SBO event as identified in Nuclear Management and Resources Council (NUMARC) 87-00, Revision 1 (Reference 53) were evaluated as part of the plant-specific analysis:

- The adequacy of the condensate/reactor coolant inventory.
- The capacity of the Class 1E batteries.
- The SBO compressed nitrogen requirements.
- The effect of loss of ventilation on rooms that contain equipment essential for plant response to an SBO event.
- The ability to maintain containment integrity.

The assessment of these characteristics determined that the plant continues to show a satisfactory response to an SBO event. Therefore, the PBAPS analysis results at TPO RTP conditions are acceptable.

Table 9-1 Key Inputs for ATWS Analysis

Parameter	CLTP/ MELLLA+	TPO RTP/ MELLLA+	Basis
Reactor Thermal Power (MWt)	3,951	4,016	[[
Analyzed Power (MWt)	3,951	4,018	
Analyzed Core Flow (Mlbm/hr / % Rated)	85.075 / 83.0	87.33 / 85.2	
Reactor Dome Pressure (psig)	1,035	1,035	
MSIV Closure Time (sec)	4.0	4.0	
High Pressure ATWS-RPT Setpoint (psig)	1,106.0	1,106.0	
Low Pressure Isolation Setpoint (psig)	825.0	825.0	
RCIC Flow Rate (gpm)	600.0	600.0	
HPCI Flow Rate (gpm)	5,000.0	5,000.0	
Number of SRVs / SRVs OOS	11 / 1 [1]	11 / 1 [1]	
Number of SSVs / SSVs OOS	3 / 0 [1]	3 / 0 [1]	
Each SRV Capacity at 1,080 psig (lbm/hr)	800,000	800,000	
SRV Analytical Opening Setpoints (psig)	1,169.1 – 1,189.7	1,169.1 – 1,189.7	
SLCS Injection Location	LP	LP	
SLCS Injection Rate (gpm)	49.1	49.1	
Boron-10 Enrichment (Atom %)	92.0	92.0	
Sodium Pentaborate Concentration (% by Weight)	8.32	8.32	
SLCS Liquid Transport Time (sec)	20.0	20.0	
Initial Suppression Pool Liquid Volume (ft^3)	122,900	122,900	
Initial Suppression Pool Temperature (°F)	86.0	86.0	
Number of RHR Suppression Pool Cooling Loops	1	1	
RHR Heat Exchanger Effectiveness per Loop (BTU/sec-°F)	610.0	610.0	
RHR Service Water Temperature (°F)	86.0	86.0	]]

Note:

1. [[ ]]

Table 9-2 Results for ATWS Analysis

ATWS Acceptance Criteria	CLTP/ MELLLA+	TPO RTP/ MELLLA+	Design Limit
Peak Vessel Bottom Pressure: TRACG (psig)	[[		1,500
Peak Suppression Pool Temperature (°F)			180.0
Peak Containment Pressure (psig)			56.0
PCT (°F) [1]			2,200
Peak Local Cladding Oxidation (%)			17
PCT: ATWSI, TRACG (°F) [1]		]]	2,200

Notes:

1. [[
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2. [[ ]]

Table 9-3 MSIVC Sequence of Events

Item	Event	TPO RTP BOC Event Time (sec)	TPO RTP EOC Event Time (sec)
1	MSIV Isolation Initiated	[[	
2	High Pressure ATWS Setpoint		
3	MSIVs Fully Closed		
4	Peak Neutron Flux		
5	Recirculation Pumps Trip		
6	Opening of the First Relief Valve		
7	Peak Heat Flux		
8	Peak Vessel Pressure		
9	Feedwater Reduction Initiated [1]		
10	SLCS Pumps Start		
11	RHR Cooling Initiated		
12	Hot Shutdown Boron Weight Achieved and Initiate Level Increase [2]		
13	Hot Shutdown Achieved (Neutron Flux Remains < 0.1%)		
14	Peak Suppression Pool Temperature		]]

Notes:

1. The feedwater pumps trip at 30 seconds after isolation.
2. (1) HSBW time = SLCS pump start + boron transportation delay + hot shutdown boron volume / SLCS flow rate * 60 sec/min;

   (2) Normal water level achieved over 200 seconds after HSBW injected.

Table 9-4 PRFO Sequence of Events

Item	Event	TPO RTP BOC Event Time (sec)	TPO RTP EOC Event Time (sec)
1	TCV and Bypass Valves Start Open	[[	
2	MSIV Closure Initiated by Low Steam Line Pressure		
3	MSIVs Fully Closed		
4	Peak Neutron Flux		
5	High Pressure ATWS Setpoint		
6	Recirculation Pumps Trip		
7	Opening of the First Relief Valve		
8	Peak Heat Flux		
9	Peak Vessel Pressure		
10	Feedwater Reduction Initiated [1]		
11	SLCS Pumps Start		
12	RHR Cooling Initiated		
13	Hot Shutdown Boron Weight Achieved and Initiate Level Increase [2]		
14	Hot Shutdown Achieved (Neutron Flux Remains < 0.1%)		
15	Peak Suppression Pool Temperature		]]

Notes:

1. The feedwater pumps trip at 30 seconds after isolation.

2. (1) HSBW time = SLCS pump start + boron transportation delay + hot shutdown boron volume / SLCS flow rate * 60 sec/min;

   (2) Normal water level achieved over 200 seconds after HSBW injected.

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Figure 9-1 TPO RTP MELLLA+ BOC PRFO (TRACG)

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Figure 9-2 TPO RTP MELLLA+ BOC MSIVC (TRACG)

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Figure 9-3 TPO RTP MELLLA+ BOC PRFO (Short-Term)

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Figure 9-4a TPO RTP MELLLA+ BOC PRFO (Long-Term)

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Figure 9-4b TPO RTP MELLLA+ BOC PRFO (Long-Term)

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Figure 9-4c TPO RTP MELLLA+ BOC PRFO (Long-Term)

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Figure 9-5 TPO RTP MELLLA+ BOC MSIVC (Short-Term)

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Figure 9-6a TPO RTP MELLLA+ BOC MSIVC (Long-Term)

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Figure 9-6b TPO RTP MELLLA+ BOC MSIVC (Long-Term)

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Figure 9-6c TPO RTP MELLLA+ BOC MSIVC (Long-Term)

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Figure 9-7 TPO RTP MELLLA+ EOC PRFO (Short-Term)

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Figure 9-8a TPO RTP MELLLA+ EOC PRFO (Long-Term)

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Figure 9-8b TPO RTP MELLLA+ EOC PRFO (Long-Term)

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Figure 9-8c TPO RTP MELLLA+ EOC PRFO (Long-Term)

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Figure 9-9 TPO RTP MELLLA+ EOC MSIVC (Short-Term)

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Figure 9-10a TPO RTP MELLLA+ EOC MSIVC (Long-Term)

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Figure 9-10b TPO RTP MELLLA+ EOC MSIVC (Long-Term)

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Figure 9-10c TPO RTP MELLLA+ EOC MSIVC (Long-Term)

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Figure 9-11 ATWS Instability – TPO RTP MELLLA+ MOC TTWBP (TRACG)

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Figure 9-12 ATWS Instability – TPO RTP MELLLA+ MOC TTWBP (TRACG)

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Figure 9-13 ATWS Instability – TPO RTP MELLLA+ MOC RPT (TRACG)

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Figure 9-14 ATWS Instability – TPO RTP MELLLA+ MOC RPT (TRACG)

10.0 OTHER EVALUATIONS

10.1 HIGH ENERGY LINE BREAK

Because the TPO uprate system operating temperatures and pressures either remain unchanged or only change slightly for high energy systems, there is no significant change in HELB mass and energy releases. The changes are insignificant in relation to the effect on line break calculations and the existing analyses bound TPO uprate conditions. Vessel dome pressure and other portions of the RCPB remain at current operating pressure or lower. The postulated break locations remain the same because the piping configuration does not change due to the TPO uprate. Therefore, the consequences of any postulated HELB remain bounded for the TPO uprate.

The HELB evaluation was performed for all systems evaluated in the UFSAR. At the TPO RTP, HELBs outside the drywell would result in an insignificant change in the sub-compartment pressure and temperature profiles. The affected building and cubicles that support safety-related functions are designed to withstand the resulting pressure and thermal loading following a HELB at the TPO RTP. A brief discussion of each break follows.

10.1.1 Steam Line Breaks

The critical parameter affecting the high energy steam line break analysis is the reactor vessel dome pressure. Because the operating pressure and flow restrictor remain unchanged, there is no change in steam line break flow rate. The MS line beak (MSLB) is used to establish the peak pressure and the temperature environment in the MS tunnel. Design margins within the HELB analysis for a MSLB provide adequate margin to the limits in the steam tunnel. With the constant pressure uprate, there is also no change in HPCI or RCIC steam line operating pressures or calculated HELB mass and energy releases with TPO uprate. Therefore, existing HELB analyses for these breaks remain bounding for TPO uprate.

10.1.2 Liquid Line Breaks

10.1.2.1 Feedwater Line Breaks

The TPO uprate affects the FW temperature by $< 2°F$ and enthalpy by less than 2.0 BTU/lbm, which results in an insignificant increase in FW mass and energy release. As a result of the small change in FW energy, the blowdown and energy release rate increase marginally. For small changes in FW process parameters, the feedwater line break conditions are bounded by the MSLB conditions in the MS tunnel. MSLB continues to be the bounding pipe break for the MS tunnel. Therefore, the original HELB analysis is bounding.

10.1.2.2 ECCS Line Breaks

ECCS liquid lines are normally isolated from the reactor during normal operations and are excluded from the PBAPS design and licensing basis for HELB. Therefore, the previous HELB analysis for breaks outside primary containment is bounding for the TPO uprate condition.

10.1.2.3 RCIC and HPCI System Line Breaks

RCIC and HPCI liquid lines are normally isolated from the reactor during normal operations and are excluded from the PBAPS design and licensing basis for HELB. Because there is no increase in the reactor dome pressure relative to the CLTP analysis, the mass and energy release for postulated RCIC and HPCI steam line breaks does not increase. Therefore, the previous HELB analysis is bounding for the TPO uprate conditions.

10.1.2.4 RWCU System Line Breaks

The existing design basis calculations bound TPO uprate conditions for evaluating the blowdown rate and energy release rate; therefore, the current HELB analyses bound the TPO uprate.

10.1.2.5 CRD System Line Breaks

The CRD system and supporting equipment operation are not affected by a TPO uprate. CRD is not considered to be a high energy system and is excluded from HELB analysis per the PBAPS design and licensing basis. Therefore, CRD lines are not affected.

10.1.2.6 Building Heating and Auxiliary Steam Line Breaks

Building heating and auxiliary steam systems are not considered to be high energy systems and are excluded from HELB analysis per the PBAPS design and licensing basis (Reference 54). Therefore, building heating and auxiliary steam lines are not affected.

10.1.2.7 Pipe Whip and Jet Impingement

Because there is no change in the nominal vessel dome pressure, pipe whip and jet impingement loads do not significantly change. Existing calculations supporting the dispositions of potential targets of pipe whip and jet impingement from postulated HELBs bound the safe shutdown effects at the TPO uprate conditions. Existing pipe whip restraints, jet impingement shields, and their supporting structures are also adequate for the TPO uprate conditions.

10.1.2.8 High Energy Sampling and Instrument Line Breaks

High energy sampling and instrument lines are determined to not be the limiting breaks at TPO uprate conditions. Therefore, high energy sampling and instrument line breaks are not affected.

10.1.2.9 Internal Flooding from HELB

None of the plant flooding zones contains a potential HELB location affected by the reactor operating conditions changed for the TPO uprate. The high energy line systems’ operational modes, plant internal flooding analysis, and safe shutdown analysis evaluated for HELB are not affected by the TPO uprate.

10.2 MODERATE ENERGY LINE BREAK

None of the plant flooding zones contains a potential moderate energy line break (MELB) location that is affected by the reactor operating conditions for the TPO uprate. The following systems contain moderate energy piping in the Reactor Building and are not affected by the TPO uprate:

- Control Rod Hydraulic,
- Residual Heat Removal,
- Standby Liquid Control,
- Reactor Core Isolation Cooling,
- Core Spray,
- Instrument Nitrogen,
- Fuel Pool Cooling,
- Post-Accident Sampling,
- High Pressure Coolant Injection,
- High Pressure Service Water,
- Emergency Service Water,
- Reactor Building Cooling Water,
- Service Air,
- Instrument Air,
- Fire Water,
- Domestic Water,
- Demineralized Water,
- Chilled Water, and
- Radiation Monitoring.

Moderate energy BOP systems which experience an increase in pressure and/or temperature due to TPO uprate include: ES, condensate, cross-around steam, and heater drains. However, these systems are located in the Turbine Building, for which no MELB analysis is performed and no elevated EQ service conditions would occur that would affect plant safe shutdown. Therefore, the TPO uprate has no effect on potential adverse effects of breaks in moderate energy lines.

No new moderate energy lines are identified from the TPO uprate. Sources of moderate energy flooding and protection requirements for safe-shutdown equipment for a postulated MELB or equipment spray are either not dependent on power level or sources are negligibly affected with no change in protection requirements. Therefore, the plant internal flooding analysis is not affected.

10.3 ENVIRONMENTAL QUALIFICATION

Safety-related electrical components must be qualified for the environment in which they operate. The TPO increase in power level increases the radiation levels experienced by equipment during normal operation and accident conditions. Because the TPO uprate does not increase the nominal vessel dome pressure, there is a very small effect on pressure and temperature conditions experienced by equipment during normal operation and accident conditions. The resulting environmental conditions are bounded by the existing environmental parameters specified for use in the EQ program.

10.3.1 Electrical Equipment

The environmental conditions for safety-related electrical equipment were reviewed to ensure that the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate.

No change is needed for the TPO uprate.

10.3.1.1 Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on DBA-LOCA conditions and their resultant temperature, pressure, humidity and radiation consequences, and includes the environments expected to exist during normal plant operation. The current accident conditions for temperature and pressure are based on analyses initiated from at least 102% of CLTP. Normal temperatures may increase slightly near the FW and reactor recirculation (RRC) lines and will be evaluated through the EQ temperature monitoring program, which tracks such information for equipment aging considerations. The current radiation levels under normal plant conditions also increase slightly. The current plant environmental envelope for radiation is not exceeded by the changes resulting from the TPO uprate.

10.3.1.2 Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from an MSLB in the steam tunnel, or other HELBs, whichever is limiting for each area. The existing HELB pressure and temperature profiles bound the TPO uprate conditions. The current plant environmental envelope for radiation is not exceeded by the changes resulting from the TPO uprate.

10.3.2 Mechanical Equipment with Non-Metallic Components

Operation at the TPO RTP level increases the normal process temperature very slightly in the FW and RRC piping. Mechanical equipment is excluded from the equipment qualification program.

10.3.3 Mechanical Component Design Qualification

The increase in power level increases the radiation levels experienced by equipment during normal operation. However, where the previous accident analyses have been based on 102% of CLTP, the accident pressures, temperatures and radiation levels do not change. The mechanical design of equipment and components (e.g., valves, heat exchangers, pumps, snubbers) in certain systems is affected by operation at the TPO RTP level because of the slightly increased temperature and, in some cases, increased flow rate. The revised operating conditions do not significantly affect the cumulative usage fatigue factors of mechanical components.

10.4 TESTING

The TPO uprate power ascension is based on the guidelines in TLTR Section L.2. Pre-operational tests are not needed because there are no significant changes to any plant systems or components that require such testing.

In preparation for operation at TPO uprate conditions, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration are taken near 95% and 100% of CLTP, and at full TPO RTP conditions. The measurements will be taken along the same rod pattern line used for the increase to TPO RTP. Core power from the APRMs is re-scaled to the TPO RTP before exceeding the CLTP and any necessary adjustments will be made to the APRM alarm and trip settings.

The turbine pressure controller setpoint will be readjusted at ≤ 95% of CLTP and held constant. The setpoint is reduced so the reactor dome pressure is the same at TPO RTP as for the CLTP. Adjustment of the pressure setpoint before taking the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the TCVs.

Demonstration of acceptable fuel thermal margin will be performed prior to and during power ascension to the TPO RTP at each steady-state heat balance point defined above. Fuel thermal margin will be projected to the TPO RTP point after the measurements taken at 95% and 100% of CLTP to show the estimated margin. The thermal margin will be confirmed by the measurements taken at full TPO RTP conditions. The demonstration of core and fuel conditions will be performed with the methods currently used at PBAPS.

Performance of the pressure and FW/level control systems will be recorded at each steady-state point defined above. The checks will utilize the methods and criteria described in the original startup testing of these systems to demonstrate acceptable operational capability. Water level changes of ±3 inches and pressure setpoint step changes of ±3 psi will be used. If necessary, adjustments will be made to the controllers and actuator elements.

Because level and pressure changes can produce power excursions above the initial condition for these tests, the final tests will be performed at a power level with a margin to TPO RTP equal to the largest anticipated excursion. The magnitude of the anticipated excursions is based on those experienced in the same tests performed at 95% and 100% of CLTP projected to TPO RTP (and other available operating experience). The intention of this margin is to avoid exceeding the licensed power limit (re: NRC RIS 2007-21, Reference 55), while creating the largest practical power difference from CLTP to obtain responses that are representative of TPO power.

The increase in power for the TPO uprate is sufficiently small that large transient tests are not necessary. High power testing performed during initial startup demonstrated the adequacy of the safety and protection systems for such large transients. Operational occurrences have shown the unit response is clearly bounded by the safety analyses for these events.

10.5 OPERATOR TRAINING AND HUMAN FACTORS

No additional training (apart from normal training for plant changes) is required to operate the plant in the TPO uprate condition. For TPO uprate conditions, operator response to transient, accident, and special events is not affected except for a negligible effect on an FSSD event, as described in Section 6.7.1. There are no required changes to the operating procedures or implementing strategy for the TPO uprate.

For TPO uprate conditions, operator response to transient, accident, and special events is not affected except for a negligible effect to an FSSD event. For FSSD Method C1, there is a small reduction in the time to initiate RPV depressurization from the control room. The reduction in time to initiate RPV depressurization is not significant because the action is completely performed in the control room and the strategy is unchanged. The reduction in time for RPV depressurization does not affect the previously NRC-approved 10 CFR 50 Appendix R exemption for a manual operator action required for low pressure ECCS injection in support of FSSD Method C1. The time required to perform the necessary actions for low pressure ECCS injection remains bounded, with sufficient margin, by the time analyzed to reach ECCS injection initiation pressure during RPV depressurization. Therefore, the time line for this existing exemption remains unchanged.

Operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the TPO uprate. There are no required changes to the operating procedures or implementing strategy for the TPO uprate.

10.6 PLANT LIFE

Three degradation mechanisms may be influenced by the TPO uprate: (1) irradiation assisted stress corrosion cracking (IASCC); (2) FAC; and (3) intergranular stress corrosion cracking (IGSCC). The increase in irradiation of the core internal components influences IASCC. The increases in steam and FW flow rate influence FAC. However, the sensitivity to the TPO uprate is small and various programs are currently implemented to monitor the aging of plant components, including EQ, FAC, and in-service inspection. EQ is addressed in Section 10.3, and FAC is addressed in Section 3.5. These programs address the degradation mechanisms and do not change for the TPO uprate. The core internals experience a slight increase in fluence, but the inspection strategy used at PBAPS, based on the BWRVIP, is sufficient to address the increase. The Maintenance Rule also provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

IGSCC is the primary degradation mechanism identified for the RCPB. For IGSCC to occur, three conditions must be present: (1) a susceptible material, (2) the presence of residual or applied tensile stress (such as from welding), and (3) a conducive environment.

Consistent with M+LTR SER Limitation and Condition 12.9, several IGSCC mitigation processes have been applied at PBAPS to reduce the RCPB components' susceptibility to IGSCC. PBAPS was designed, fabricated, and constructed with most welds either using corrosion resistant materials, solution treated, or clad with resistant materials. Stress improvement processes and original construction processes used for IGSCC resistance are not affected by TPO.

In addition, PBAPS has implemented hydrogen water chemistry with noble metals consistent with EPRI water chemistry guidelines. This reduces the susceptibility of materials exposed to reactor coolant thus improving resistance to stress corrosion cracking. The change to temperature and flow conditions for portions of the RCPB piping from TPO operation does not affect the other susceptibility factors associated with IGSCC. The three conditions for IGSCC remain unchanged under TPO operation. Therefore, implementation of TPO has a negligible effect on IGSCC potential.

The longevity of most equipment is not affected by the TPO uprate because there is no significant change in the operating conditions and any changes in operating conditions as a result of TPO are bounded by a previous evaluation completed for EPU at 102% of CLTP. No additional maintenance, inspection, testing, or surveillance procedures are required.

10.7 NRC and Industry Communications

NRC and industry communications are addressed in TLTR, Section 10.8. In accordance with the TLTR, it is not necessary to review prior dispositions of NRC and industry communications and no additional information is required in this area.

10.8 Plant Procedures and Programs

PBAPS has previously implemented a TPO uprate, including procedure and program requirements. Therefore, only minor changes are required to restore the TPO requirements to ensure that plant procedures and programs are in place to:

1. Monitor and maintain instrument calibration during normal plant operation to ensure that instrument uncertainty is not greater than the uncertainty used to justify the TPO uprate;
2. Control the software and hardware configuration of the associated instrumentation;
3. Perform corrective actions, where required, to maintain instrument uncertainty within limits;
4. Report deficiencies of the associated instruments to the manufacturer; and
5. Receive and resolve the manufacturer’s deficiency reports.

10.9 Emergency Operating Procedures

The EOP action thresholds are plant unique and will be addressed using standard procedure updating processes. The TPO uprate will have no effect on the EOP strategies and only minor changes to operator action thresholds.

10.10 Individual Plant Examination

PBAPS maintains and regularly updates a station PRA model. Use of the model is integrated with station operations and decision-making.

The PBAPS IPE PRA model and analysis will not be specifically updated for TPO because the change in plant risk from the TPO uprate is insignificant. This conclusion is supported by NRC RIS 2002-03 (Reference 17). In response to feedback received during the public workshop held on August 23, 2001, the NRC wrote, “The NRC has generically determined that measurement uncertainty recapture power uprates have an insignificant effect on plant risk. Therefore, no risk information is requested to support such applications.” (Reference 17).

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Appendix A – Limitations from Safety Evaluation for LTR NEDC-33173P

Disposition of additional limitations and conditions related to the SE for NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains"

There are 24 limitations and conditions listed in Section 9 of the Methods LTR SER (Reference 12). The table below lists each of the 24 limitations and conditions and identifies which section of the TSAR discusses compliance with each limitation and condition.

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.1	TGBLA/PANAC Version	The neutronic methods used to simulate the reactor core response and that feed into the downstream safety analyses supporting operation at EPU/MELLLA+ will apply TGBLA06/PANAC11 or later NRC-approved version of neutronic method.	Comply	Table 1-1a and Section 2.6.1
9.2	3D Monicore	For EPU/MELLLA+ applications, relying on TGBLA04/PANAC10 methods, the bundle RMS difference uncertainty will be established from plant-specific core-tracking data, based on TGBLA04/PANAC10. The use of plant-specific trendline based on the neutronic method employed will capture the actual bundle power uncertainty of the core monitoring system.	N/A	(1)
9.3	Power/Flow Ratio	Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.	Comply	Sections 1.3.1 and 2.2.5 (2)
9.4	SLMCPR 1	Limitation has been removed according to Appendix I of this SE.	N/A	(3)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.5	SLMCPR 2	This Limitation has been revised according to Appendix I of this SE. For operation at MELLLA+, including operation at the EPU power levels at the achievable core flow state-point, a 0.01 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios up to 42 MWt/Mlbm/hr, and a 0.02 value shall be added to the cycle-specific SLMCPR value for power-to-flow ratios above 42 MWt/Mlbm/hr.	Comply	Sections 2.2.1 and 2.2.5
9.6	R-Factor	The plant-specific R-factor calculation at a bundle level will be consistent with lattice axial void conditions expected for the hot channel operating state. The plant-specific EPU/MELLLA+ application will confirm that the R-factor calculation is consistent with the hot channel axial void conditions.	Comply	Section 2.2
9.7	ECCS-LOCA 1	For applications requesting implementation of EPU or expanded operating domains, including MELLLA+, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.	Comply	Section 4.3 M+SAR Sections 4.3.2 and 4.3.3 (4) (5) (6)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.8	ECCS-LOCA 2	The ECCS-LOCA will be performed for all statepoints in the upper boundary of the expanded operating domain, including the minimum core flow statepoints, the transition statepoint as defined in Reference 11 and the 55 percent core flow statepoint. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The SRLR will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.	Comply	Section 4.3 M+SAR Section 4.3.3 (2) (6) (7)
9.9	Transient LHGR 1	Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the thermal-mechanical (T-M) acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for the both the UO_2 and the limiting GdO_2 *[sic]* rods.	Comply	(8)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.10	Transient LHGR 2	Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR or COLR, or it will be reported directly to the NRC as an attachment to the SRLR or COLR.	Comply	(8)
9.11	Transient LHGR 3	To account for the impact of the void history bias, plant-specific EPU and MELLLA+ applications using either TRACG or ODYN will demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for all of limiting AOO transient events, including equipment out-of-service. Limiting transients in this case, refers to transients where the void reactivity coefficient plays a significant role (such as pressurization events). If the void history bias is incorporated into the transient model within the code, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain is no longer required.	Comply	(8)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.12	LHGR and Exposure Qualification	In MFN 06-481, GE committed to submit plenum fission gas and fuel exposure gamma scans as part of the revision to the T-M licensing process. The conclusions of the plenum fission gas and fuel exposure gamma scans of GE 10x10 fuel designs as operated will be submitted for NRC staff review and approval. This revision will be accomplished through Amendment to GESTAR II or in a T-M licensing LTR. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review (Reference 9). Once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P must utilize the PRIME T-M methods.	Comply	Section 2.6.3 (9)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.13	Application of 10 Weight Percent Gd	Before applying 10 weight percent Gd to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for thermal overpower (TOP) and mechanical overpower (MOP) conditions that bounds the response of plants operating at EPU and expanded operating domains at the most limiting statepoints, considering the operating flexibilities (e.g., equipment out-of-service). Before the use of 10 weight percent Gd for modern fuel designs, NRC must review and approve TGBLA06 qualification submittal. Where a fuel design refers to a design with Gd-bearing rods adjacent to vanished or water rods, the submittal should include specific information regarding acceptance criteria for the qualification and address any downstream impacts in terms of the safety analysis. The 10 weight percent Gd qualifications submittal can supplement this report.	N/A	(10)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.14	Part 21 Evaluation of GESTR-M Fuel Temperature Calculation	Any conclusions drawn from the NRC staff evaluation of the GE's Part 21 report will be applicable to the GESTR-M T-M assessment of this SE for future license application. GE submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform GE of its conclusions.	N/A	(11)
9.15	Void Reactivity 1	The void reactivity coefficient bias and uncertainties in TRACG for EPU and MELLLA+ must be representative of the lattice designs of the fuel loaded in the core.	Comply	Section 2.2 and M+SAR Section 9.1.1 (12)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.16	Void Reactivity 2	A supplement to TRACG/PANAC11 for AOO is under NRC staff review (Reference 5). TRACG internally models the response surface for the void coefficient biases and uncertainties for known dependencies due to the relative moderator density and exposure on nodal basis. Therefore, the void history bias determined through the methods review can be incorporated into the response surface "known" bias or through changes in lattice physics/core simulator methods for establishing the instantaneous cross-sections. Including the bias in the calculations negates the need for ensuring that plant-specific applications show sufficient margin. For application of TRACG to EPU and MELLLA+ applications, the TRACG methodology must incorporate the void history bias. The manner in which this void history bias is accounted for will be established by the NRC staff SE approving NEDE-32906P, Supplement 3, "Migration to TRACG04/ PANAC11 from TRACG02/PANAC10," May 2006 (Reference 13). This limitation applies until the new TRACG/PANAC methodology is approved by the NRC staff.	N/A	(13)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.17	Steady-State 5 Percent Bypass Voiding	The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (LRPM levels). Limiting the bypass voiding to less than 5 percent for long-term steady operation ensures that instrumentation is operated within the specification. For EPU and MELLLA+ operation, the bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions within the MELLLA+ upper boundary. The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.	Comply	Sections 2.1.2 and 5.1.1.2 (2)
9.18	Stability Setpoints Adjustment	The NRC staff concludes that the presence bypass voiding at the low-flow conditions where instabilities are likely can result in calibration errors of less than 5 percent for OPRM cells and less than 2 percent for APRM signals. These calibration errors must be accounted for while determining the setpoints for any detect and suppress long term methodology. The calibration values for the different long-term solutions are specified in the associated sections of this SE, discussing the stability methodology.	N/A	Section 2.4.1 (14)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.19	Void-Quality Correlation 1	For applications involving PANCEA/ODYN/ISCOR/ TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady-state, transient, and accident conditions.	N/A	Section 2.2.2 (2) (15)
9.20	Void-Quality Correlation 2	The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, "Migration to TRACG04/ PANAC11 from TRACG02/ PANAC10," dated May 2006 (Reference 13). The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC-32906P (Reference 13) will be applicable as approved.	Comply	Section 9.3.1 (16)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.21	Mixed Core Method 1	Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise LTR NEDC-33173P (Reference 12) for mixed core application.	N/A	(17)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.22	Mixed Core Method 2	For any plant-specific applications of TGBLA06 with fuel type characteristics not covered in this review, GE needs to provide assessment data similar to that provided for the GEH/GNF fuels. The Interim Methods review is applicable to all GEH/GNF lattices up to GNF2. Fuel lattice designs, other than GEH/GNF lattices up to GNF2, with the following characteristics are not covered by this review: • square internal water channels water crosses • Gd rods simultaneously adjacent to water and vanished rods • 11x11 lattices • MOX fuel The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains. Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GEH methods may be applied.	N/A	(17)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.23	MELLLA+ Eigenvalue Tracking	In the first plant-specific implementation of MELLLA+, the cycle-specific eigenvalue tracking data will be evaluated and submitted to NRC to establish the performance of nuclear methods under the operation in the new operating domain. The following data will be analyzed: • Hot critical eigenvalue, • Cold critical eigenvalue, • Nodal power distribution (measured and calculated TIP comparison), • Bundle power distribution (measured and calculated TIP comparison), • Thermal margin, • Core flow and pressure drop uncertainties, and • The MCPR importance parameter (MIP) Criterion (e.g., determine if core and fuel design selected is expected to produce a plant response outside the prior experience base). Provision of evaluation of the core-tracking data will provide the NRC staff with bases to establish if operation at the expanded operating domain indicates: (1) changes in the performance of nuclear methods outside the EPU experience base; (2) changes in the available thermal margins; (3) need for changes in the uncertainties and NRC-approved criterion used in the SLMCPR methodology; or (4) any anomaly that may require corrective actions.	Comply	(18)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
9.24	Plant-Specific Application	The plant-specific applications will provide prediction of key parameters for cycle exposures for operation at EPU (and MELLLA+ for MELLLA+ applications). The plant-specific prediction of these key parameters will be plotted against the EPU Reference Plant experience base and MELLLA+ operating experience, if available. For evaluation of the margins available in the fuel design limits, plant-specific applications will also provide quarter core map (assuming core symmetry) showing bundle power, bundle operating LHGR, and MCPR for BOC, MOC, and EOC. Because the minimum margins to specific limits may occur at exposures other than the traditional BOC, MOC, and EOC, the data will be provided at these exposures.	Comply	Section 2.1.2

Notes:

1. As shown in Table 1-1a, the PBAPS M+SAR and PBAPS TSAR are based on TGBLA06/PANAC11, not TGBLA 04/PANAC10.
2. Correspondence concerning implementation of this limitation and condition is docketed in the letter from James F. Harrison (GEH) to NRC, "Implementation of Methods Limitations - NEDC-33173P," MFN 08-693, September 18, 2008 (Reference 12).
3. This limitation was removed as noted in Reference 12.
4. The PBAPS M+SAR (Reference 15), Sections 4.3.2 and 4.3.3 confirm that the conclusions based on licensing basis PCT are also valid for the Upper Bound PCT.
5. Top-peaked and mid-peaked cases are reported for each break type, using the bounding shape, to determine the licensing basis peak cladding temperature (LBPCT) and upper bound peak cladding temperature (UBPCT). LBPCT and UBPCT are reported for small break only as the large break case is not limiting; only one LBPCT is reported per Reference 11.
6. ECCS-LOCA analyses are not affected by TPO, because the evaluations were already performed at 102% of CLTP in the M+SAR. No new analysis is required as stated in TLTR SER (Reference 1). The existing M+SAR ECCS-LOCA analysis continues to bound the TSAR analysis.
7. The minimum CF and low flow (55%) cases are reported to demonstrate the limiting result along the upper boundary. For large break cases, the transition point is dispositioned by evaluation because the large break is shown to be bounded by the small break result.
8. As discussed in the PBAPS M+SAR (Reference 15), Section 9.1.1, fuel rod T-M performance will be evaluated as part of the reload licensing analyses performed for the cycle specific core. [[
 ]]
9. The PRIME LTR and its application (Reference 9) was approved on January 22, 2010 and implemented in GESTAR II (Reference 16) in September 2010. The PBAPS M+SAR and PBAPS TSAR are based on the GNF2 fuel product line, which has a PRIME T-M basis. PRIME fuel parameters will be used in all analyses requiring fuel performance parameters.
10. PBAPS M+SAR and PBAPS TSAR use GNF2 fuel, and as such does not seek to apply 10 wt% Gd to this licensing application.
11. This limitation and condition relates to GEH's treatment of the NRC staff review of the 10 CFR 21 report related to the GESTR-M T-M evaluation. The PBAPS M+SAR and PBAPS TSAR are based on the GNF2 fuel product line, which has a PRIME T-M and PRIME fuel temperature basis included. Therefore, this limitation is no longer applicable.
12. The PBAPS M+SAR and PBAPS TSAR licensing basis use TRACG for AOO, DSS-CD, ATWS overpressure, and ATWSI analyses. The void reactivity coefficients bias and uncertainties used in the latest version of TRACG are in accordance with Reference 13 and are applicable to the GNF2 lattice designs loaded in the core.

13. The PBAPS TSAR licensing basis ATWS overpressure and ATWSI analyses use the TRACG code. The void reactivity coefficients bias and uncertainties used in the latest version of TRACG are applicable to the GNF2 lattice designs loaded in the core. The PBAPS TSAR licensing basis ATWS PCT, suppression pool temperature, and containment pressure analyses use the ODYN code.
14. Not applicable to DSS-CD because the significant conservatisms in the current licensing methodology and associated MCPR margins are more than sufficient to compensate for the overall uncertainty in the OPRM instrumentation.
15. The limiting fuel thermal margin transients for the PBAPS M+SAR and PBAPS TSAR are determined using a plant-specific TRACG model that is compliant with Reference 13. The NRC SE for Reference 13 states that this 0.01 OLMCPR penalty is not applicable to analysis using a TRACG model compliant with Reference 13. Therefore, this commitment to add an additional 0.01 penalty to the calculated OLMCPR is not applicable to PBAPS.
16. The PBAPS TSAR licensing basis uses TRACG for ATWS overpressure and ATWSI analyses. The interfacial shear model used in the latest version of TRACG is applicable to the GNF2 design loaded in the core.
17. The PBAPS M+SAR and PBAPS TSAR are based on a GNF2 equilibrium core design. Therefore, the mixed core limitations are not applicable.
18. Correspondence concerning implementation of this limitation and condition is docketed in the letter from James F. Harrison (GEH) to NRC, “Clarification of Limitation and Condition 23 for NEDC-33173P, ‘Applicability of GE Methods to Expanded Operating Domains’,” MFN 15-066, August 26, 2015 (Reference 12).

Appendix B - Limitations from Safety Evaluation for LTR NEDC-33006P

Disposition of additional limitations and conditions related to the SE for

NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus”

There are 54 limitations and conditions listed in Section 12 of the M+LTR SER (Reference 11). The table below lists each of the 54 limitations and conditions and identifies which section of the M+SAR discusses compliance with each limitation and condition.

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.1	GEXL-PLUS	The plant-specific application will confirm that for operation within the boundary defined by the MELLLA+ upper boundary and maximum CF range, the GEXL-PLUS experimental database covers the thermal-hydraulic conditions the fuel bundles will experience, including, bundle power, mass flux, void fraction, pressure, and subcooling. If the GEXL-PLUS experimental database does not cover the within bundle thermal-hydraulic conditions, during steady state, transient conditions, and DBA conditions, GHNE will inform the NRC at the time of submittal and obtain the necessary data for the submittal of the plant-specific MELLLA+ application. In addition, the plant-specific application will confirm that the experimental pressure drop database for the pressure drop correlation covers the pressure drops anticipated in the MELLLA+ range. With subsequent fuel designs, the plant-specific applications will confirm that the database supporting the CPR correlations covers the powers, flows and void fractions BWR bundles will experience for operation at and within the MELLLA+ domain, during steady state, transient, and DBA conditions. The plant-specific submittal will also confirm that the NRC staff reviewed and approved the associated CPR correlation if the changes in the correlation are outside the GESTAR II (Amendment 22) process. Similarly, the plant-specific application will confirm that the experimental pressure drop database does cover the range of pressures the fuel bundles will experience for operation within the MELLLA+ domain.	Comply	Section 2.6.4 M+SAR Sections 1.1.3 and 2.6.4

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.2	Related LTRs	Plant-specific MELLLA+ applications must comply with the limitations and conditions specified in and be consistent with the purpose and content covered in the NRC staff SEs approving the latest version of the following LTRs: NEDC-33173P, NEDC-33075P-A, and NEDC-33147-A.	Comply	Section 1.2.1
12.3.a	Concurrent Changes	The plant-specific analyses supporting MELLLA+ operation will include all operating condition changes that are implemented at the plant at the time of MELLLA+ implementation. Operating condition changes include, but are not limited to, those changes that affect, an increase in the dome pressure, maximum CF, fuel cycle length, or any changes in the licensed operational enhancements. For example, with an increase in dome pressure, the following analyses must be analyzed: the ATWS analysis, the ASME overpressure analyses, the transient analyses, and the ECCS-LOCA analysis. Any changes to the safety system settings or any actuation setpoint changes necessary to operate with the increased dome pressure must be included in the evaluations (e.g., SRV setpoints).	Comply	Section 1.2.3

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.3.b		For all topics in LTR NEDC-33006P that are reduced in scope or generically dispositioned, the plant-specific application will provide justification that the reduced scope or generic disposition is applicable to the plant. If changes that invalidate the LTR dispositions are to be implemented at the time of MELLLA+ implementation, the plant-specific application will provide analyses and evaluations that demonstrate the cumulative effect with MELLLA+ operation. For example, if the dome pressure is increased, the ECCS performance will be evaluated on a plant-specific basis.	Comply	(1)
12.3.c		Any generic bounding sensitivity analyses provided in LTR NEDC-33006P will be evaluated to ensure that the key plant-specific input parameters and assumptions are applicable and bounded. If these generic sensitivity analyses are not applicable or additional operating condition changes affect the generic sensitivity analyses, a plant-specific evaluation will be provided. For example, with an increase in the dome pressure, the ATWS sensitivity analyses that model operator actions (e.g., depressurization if the HCTL is reached) needs to be reanalyzed, using the bounding dome pressure condition.	Comply	(1)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.3.d		If a new GE fuel product line or another vendor's fuel is loaded at the plant, the applicability of any generic sensitivity analyses supporting the MELLLA+ application shall be justified in the plant-specific application. If the generic sensitivity analyses cannot be demonstrated to be applicable, the analyses will be performed including the new fuel. For example, the ATWS instability analyses supporting the MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS instability performance of the new GE fuel or another vendor's fuel for MELLLA+ operation shall be provided to support the plant-specific application.	Comply	Section 9.3.1.4 (2)
12.3.e		If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the analyses supporting the plant-specific MELLLA+ application will be based on a specific core configuration or bounding core conditions. Any topics that are generically dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration or bounding core conditions will be provided.	Comply	Section 2.1.1

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.3.f		If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the plant-specific application will reference an NRC-approved stability method supporting MELLLA+ operation, or provide sufficient plant-specific information to allow the NRC staff to review and approve the stability method supporting MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluations supporting the stability method are applicable to the fuel loaded in the core.	Comply	Section 2.4.1
12.3.g		For MELLLA+ operation, core instability is possible in the event a transient or plant maneuver places the reactor at a high power/low-flow condition. Therefore, plants operating at MELLLA+ conditions must have a NRC-approved instability protection method. In the event the instability protection method is inoperable, the applicant must employ an NRC-approved backup instability method. The licensee will provide technical specification (TS) changes that specify the instability method operability requirements for MELLLA+ operation, including any backup stability protection methods.	Comply	Section 2.4.4

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.4	Reload Analysis Submittal	The plant-specific MELLLA+ application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel and cycle-dependent analyses including the plant-specific thermal limits assessment may be submitted by supplementing the initial M+SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.	Comply	Sections 1.2.1 and 1.2.3, Item 2.0
12.5.a	Operating Flexibility	The licensee will amend the TS limiting condition for operation (LCO) for any equipment out-of-service (i.e., SLO) or operating flexibilities prohibited in the plant-specific MELLLA+ application.	Comply	Section 1.3.2
12.5.b		For an operating flexibility, such as FWHOOS, that is prohibited in the MELLLA+ plant-specific application but is not included in the TS LCO, the licensee will propose and implement a license condition.	Comply	Section 1.3.2
12.5.c		The power flow map is not specified in the TS; however, it is an important licensed operating domain. Licensees may elect to be licensed and operate the plant under plant-specific-expanded domain that is bounded by the MELLLA+ upper boundary. Plant-specific applications approved for operation within the MELLLA+ domain will include the plant-specific power/flow map specifying the licensed domain in the COLR.	Comply	Section 1.3.1 M+SAR Section 1.2.1

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.6	SLMCPR Statepoints and CF Uncertainty	Until such time when the SLMCPR methodology (References 8 and 56) for off-rated SLMCPR calculation is approved by the staff for MELLLA+ operation, the SLMCPR will be calculated at the rated statepoint (120 percent P/100 percent CF), the plant-specific minimum CF statepoint (e.g., 120 percent P/80 percent CF), and at the 100 percent OLTP at 55 percent CF statepoint. The currently approved off-rated CF uncertainty will be used for the minimum CF and 55 percent CF statepoints. The uncertainty must be consistent with the CF uncertainty currently applied to the SLO operation or as NRC-approved for MELLLA+ operation. The calculated values will be documented in the SRLR.	Comply	Section 2.2.1
12.7	Stability	Manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. If the primary stability protection system is declared inoperable, a non-manual NRC-approved backup protection system must be provided, or the reactor core must be operated below a NRC-approved backup stability boundary specifically approved for MELLLA+ operation for the stability option employed.	Comply	Section 2.4.4
12.8	Fluence Methodology and Fracture Toughness	The applicant is to provide a plant-specific evaluation of the MELLLA+ RPV fluence using the most up-to-date NRC-approved fluence methodology. This fluence will then be used to provide a plant-specific evaluation of the RPV fracture toughness in accordance with RG 1.99, Revision 2.	Comply	Section 3.2.1

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.9	Reactor Coolant Pressure Boundary	MELLLA+ applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the MELLLA+ operation on a plant-specific basis.	Comply	Section 10.6 (3)
12.10.a	ECCS-LOCA Off-rated Multiplier	The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.	Comply	Section 4.3 and M+SAR Section 4.3.2 (4) (5)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.10.b		LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.	Comply	Section 4.3 and M+SAR Section 4.3.2 (5) (6)
12.10.c		Off-rated limits will not be applied to the minimum CF statepoint.	Comply	Section 4.3 and M+SAR Section 4.3.2 (5) (7)
12.10.d		If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.	Comply	Section 4.3 and M+SAR Section 4.3.2 (5)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.11	ECCS-LOCA Axial Power Distribution Evaluation	For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.	Comply	Section 4.3 and M+SAR Sections 4.3.2 and 4.3.3 (5) (8)
12.12.a	ECCS-LOCA Reporting	Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and	Comply	Section 4.3 and M+SAR Section 4.3.3 (4) (5)
12.12.b		The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.	Comply	Section 4.3 and M+SAR Section 4.3.3 (4) (5)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.13	Small Break LOCA	Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to [[ ]] relative to the Appendix K or the licensing basis PCT.	Comply	Section 4.3 and M+SAR Section 4.3.3 (5) (9)
12.14	Break Spectrum	The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.	Comply	Section 4.3 and M+SAR Section 4.3.1 (5)
12.15	Bypass Voiding Above the D-Level	Plant-specific MELLLA+ applications shall identify where in the MELLLA+ upper boundary the bypass voiding greater than 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.	Comply	Section 5.1.1.2

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.16	RWE	Plants operating at the MELLLA+ operating domain shall perform RWE analyses to confirm the adequacy of the generic RBM setpoints. The M+SAR shall provide a discussion of the analyses performed and the results.	Comply	Section 5.3.9
12.17	ATWS LOOP	As specified in LTR NEDC-33006P, at least two plant-specific ATWS calculations must be performed: MSIVC and PRFO. In addition, if RHR capability is affected by LOOP, then a third plant-specific ATWS calculation must be performed that includes the reduced RHR capability. To evaluate the effect of reduced RHR capacity during LOOP, the plant-specific ATWS calculation must be performed for a sufficiently large period of time after HSBW injection is complete to guarantee that the suppression pool temperature is cooling, indicating that the RHR capacity is greater than the decay heat generation. The plant-specific application should include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH. The plant-specific application should include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH.	Comply	Section 9.3.1.2 Section 4.2.5

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.18.a	ATWS TRACG Analysis	For plants that do not achieve hot shutdown prior to reaching the heat capacity temperature limit (HCTL) based on the licensing ODYN code calculation, plant-specific MELLLA+ implementations must perform best-estimate TRACG calculations on a plant-specific basis. The TRACG analysis will account for all plant parameters, including water-level control strategy and all plant-specific emergency operating procedure (EOP) actions.	N/A	Section 9.3.1 (10)
12.18.b		The TRACG calculation is not required if the plant increases the boron-10 concentration/enrichment so that the integrated heat load to containment calculated by the licensing ODYN calculation does not change with respect to a reference OLTP/75 percent flow ODYN calculation.	N/A	(10)
12.18.c		Peak cladding temperature (PCT) for both phases of the transient (initial overpressure and emergency depressurization) must be evaluated on a plant-specific basis with the TRACG ATWS calculation.	N/A	(10)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.18.d		In general, the plant-specific application will ensure that operation in the MELLLA+ domain is consistent with the assumptions used in the ATWS analysis, including equipment out of service (e.g., FWHOOS, SLO, SRVs, standby liquid control (SLC) pumps, and RHR pumps, etc.). If assumptions are not satisfied, operation in MELLLA+ is not allowed. The SRLR will specify the prohibited flexibility options for plant-specific MELLLA+ operation, where applicable. For key input parameters, systems and engineering safety features that are important to simulating the ATWS analysis and are specified in the Technical Specification (TS) (e.g., SLCS parameters, ATWS RPT, etc.), the calculation assumptions must be consistent with the allowed TS values and the allowed plant configuration. If the analyses deviate from the allowed TS configuration for long term equipment out of service (i.e., beyond the TS LCO), the plant-specific application will specify and justify the deviation. In addition, the licensee must ensure that all operability requirements are met (e.g., NPSH) by equipment assumed operable in the calculations.	Comply	(10) Section 4.2.5

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.18.e		Nominal input parameters can be used in the ATWS analyses provided the uncertainty treatment and selection of the values of these input parameters are consistent with the input methods used in the original GE ATWS analyses in NEDE-24222. Treatment of key input parameters in terms of uncertainties applied or plant-specific TS value used can differ from the original NEDE-24222 approach, provided the manner in which it is used yields more conservative ATWS results.	N/A	(10)
12.18.f		The plant-specific application will include tabulation and discussion of the key input parameters and the associated uncertainty treatment.	N/A	(10)
12.19	Plant-Specific ATWS Instability	Until such time that NRC approves a generic solution for ATWS instability calculations for MELLLA+ operation, each plant-specific MELLLA+ application must provide ATWS instability analysis that satisfies the ATWS acceptance criteria listed in SRP Section 15.8. The plant-specific ATWS instability calculation must: (1) be based on the peak-reactivity exposure conditions, (2) model the plant-specific configuration important to ATWS instability response including mixed core, if applicable, and (3) use the regional-mode nodalization scheme. In order to improve the fidelity of the analyses, the plant-specific calculations should be based on latest NRC-approved neutronic and thermal-hydraulic codes such as TGBLA06/PANAC11 and TRACG04.	Comply	Section 9.3.1.4

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.20	Generic ATWS Instability	Once the generic solution is approved, the plant-specific applications must provide confirmation that the generic instability analyses are relevant and applicable to their plant. Applicability confirmation includes review of any differences in plant design or operation that will result in significantly lower stability margins during ATWS such as: • turbine bypass capacity, • fraction of steam-driven feedwater pumps, • any changes in plant design or operation that will significantly increase core inlet subcooling during ATWS events, • significant differences in radial and axial power distributions, • hot-channel power-to-flow ratio, • fuel design changes beyond GE14.	N/A	(2)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.21	Individual Plant Evaluation	Licensees that submit a MELLLA+ application should address the plant-specific risk impacts associated with MELLLA+ implementation, consistent with approved guidance documents (e.g., NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A) and the Matrix 13 of RS-001 and re-address the plant-specific risk impacts consistent with the approved guidance documents that were used in their approved EPU application and Matrix 13 of RS-001. If an EPU and MELLLA+ application come to the NRC in parallel, the expectation is that the EPU submittal will have incorporated the MELLLA+ impacts.	Comply	Section 10.10 (11)
12.22	IASCC	The applicant is to provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes the components that will exceed the IASCC threshold of $5x10^{20}$ n/cm^2 (E>1MeV), the impact of failure of these components on the integrity of the reactor internals and core support structures under licensing design bases conditions, and the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.	Comply	Section 10.6 (12)

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.23.1	Limitations from the ATWS RAI Evaluations	See limitation 12.18.d.	Comply	Section 9.3.1.2 Section 4.2.5
12.23.2		The plant-specific ODYN and TRACG key calculation parameters must be provided to the staff so they can verify that all plant-specific automatic settings are modeled properly.	Comply	Section 9.3.1
12.23.3		The ATWS peak pressure response would be dependent upon SRVs upper tolerances assumed in the calculations. For each individual SRV, the tolerances used in the analysis must be consistent with or bound the plant-specific SRV performance. The SRV tolerance test data would be statistically treated using the NRC's historical 95/95 approach or any new NRC-approved statistical treatment method. In the event that current EPU experience base shows propensity for valve drift higher than pre-EPU experience base, the plant-specific transient and ATWS analyses would be based on the higher tolerances or justify the reason why the propensity for the higher drift is not applicable the plant's SRVs.	Comply	Section 9.3.1
12.23.4		Emergency Procedure Guideline (EPG)/Severe Accident Guideline (SAG) parameters must be reviewed for applicability to MELLLA+ operation in a plant-specific basis. The plant-specific MELLLA+ application will include a section that discusses the plant-specific EOPs and confirms that the ATWS calculation is consistent with the operator actions.	Comply	Section 9.3.1.2

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.23.5		The conclusions of this LTR and associated SE are limited to reactors operating with a power density lower than 52.5 MW/MLBM/hr for operation at the minimum allowable CF at 120 percent OLTP. Verification that reactor operation will be maintained below this analysis limit must be performed for all plant-specific applications.	Comply	Section 9.3.1.4
12.23.6		For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.	Comply	Section 9.3.1.4
12.23.7		See limitation 12.23.6.	Comply	Section 9.3.1.4
12.23.8		The plant-specific ATWS calculations must account for all plant- and fuel-design-specific features, such as the debris filters.	Comply	Section 9.3.1

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.23.9		Plant-specific applications must review the safety system specifications to ensure that all of the assumptions used for the ATWS SE indeed apply to their plant-specific conditions. The NRC staff review will give special attention to crucial safety systems like HPCI, and physical limitations like NPSH and maximum vessel pressure that RCIC and HPCI can inject. The plant-specific application will include a discussion on the licensing bases of the plant in terms of NPSH and system performance. It will also include NPSH and system performance evaluation for the duration of the event.	Comply	Section 9.3.1.5 Section 4.2.5
12.23.10		Plant-specific applications must ensure that an increase in containment pressure resulting from ATWS events with EPU/MELLLA+ operation does not affect adversely the operation of safety-grade equipment.	Comply	Section 9.3.1.2 PBAPS does not credit CAP for ECCS NPSH.
12.23.11		The plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODYN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.	Comply	Section 9.3.1.2

Limitation and Condition Number from NRC SER	Limitation and Condition Title	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
12.24.1	Limitations from Fuel Dependent Analyses RAI Evaluations	For EPU/MELLLA+ plant-specific applications that use TRACG or any code that has the capability to model in-channel water rod flow, the supporting analysis will use the actual flow configuration.	Comply	Section 2.6.2 Section 9.3.1.4 (13)
12.24.2		The EPU/MELLLA+ application would provide the exit void fraction of the high-powered bundles in the comparison between the EPU/MELLLA+ and the pre-MELLLA+ conditions.	Comply	Section 2.1.2
12.24.3		See limitation 12.6.	Comply	Section 2.2.1
12.24.4		See limitation 12.18.d.	Comply	Section 9.3.1 Section 4.2.5

Notes:

1. Section 1.1.1 of the PBAPS M+SAR (Reference 15) discusses generic assessments for the PBAPS MELLLA+ application, which demonstrates compliance with M+LTR SER Limitations and Conditions 12.3.b and 12.3.c. This TSAR relies on no generic dispositions discussed in the TLTR (Reference 1); thus, compliance is maintained.

2. This requirement relates to implementation of a generic ATWSI solution, which is not yet approved by the NRC. PBAPS at TPO and MELLLA+ conditions is based on a plant-specific ATWSI analysis.

3. Section 3.5.1.4 of the PBAPS M+SAR (Reference 15) addresses other-than-category "A" RCPB material. Because the three conditions for IGSCC remain unchanged under TPO operation (see Section 10.6 of this TSAR), the conclusions of Section 3.5.1.4 of the PBAPS M+SAR remain valid at TPO uprate conditions.

4. The analysis has shown the small break to be limiting. In accordance with the Reference 11 requirement, which is implemented by Reference 56, the survey of cases to identify the limiting power and flow statepoint has been done on an Appendix K basis, exclusively; the nominal assumption calculation is performed for the limiting case of each break size only. Consideration of the transition point for applicable large breaks has been evaluated and justified as non-limiting, with an effect sufficiently small to justify that the UBPCT and LBPCT provided is, in fact, the limiting PCT. Sections 4.3.2 and 4.3.3 confirm that conclusions based on the LBPCT are consistently valid for the UBPCT as well. With the Reference 1 requirement and the availability of nominal assumption calculations for limiting cases only, the UBPCT and the LBPCT are reported and used for defining the LBPCT with no change in applied plant variables and uncertainties.

5. ECCS-LOCA analyses are not affected by TPO, because the evaluations were already performed at 102% of CLTP in the M+SAR. No new analysis is required as stated in the TLTR SER (Reference 1). The existing M+SAR ECCS-LOCA analysis continues to bound the TSAR analysis.

6. The reload evaluation process includes a review of the cycle-specific off-rated limits and either confirms continuing applicability of the analysis basis or requires resolution for any indicated change.

7. PBAPS takes credit for off-rated limits along the MELLLA+ boundary down to the low flow point, but these begin to be applied below the MELLLA+ minimum flow point. The analysis complies with the requirement to not apply off-rated limits at the minimum CF point.

8. Top-peaked and mid-peaked cases are reported for each break type, using the bounding shape to determine the LBPCT and the UBPCT. The LBPCT and UBPCT are reported only for the small break because the large break case is not limiting; therefore, only one LBPCT is reported, per Reference 11.

9. This limitation and condition is written as a check against a potentially limiting small break case going undetected considering bounding large break calculated results. The analysis for

PBAPS MELLLA+ is small break limited. The bounding result at the minimum CF statepoint is calculated and reported. Because it is not limiting based on the small break LOCA analysis performed at the rated (flow) CLTP condition (Criteria 1), and the margin boundary of 50ºF is not relevant because the small break is bounding (Criteria 2), the calculation of the transition statepoint is not required. The transition statepoint is relevant to the large break case as discussed in Reference 11. Section 4.3.2.3 of Reference 12 confirms that for small breaks, a decrease in the power will reduce the PCT much more than any flow reduction; therefore, the small break transition point does not challenge the determination of limiting PCT and is not reported.

10. As further discussed in Section 9.3.1, the best-estimate ATWS with emergency depressurization analysis is not required as the peak suppression pool temperature remains below the HCTL curves.

11. Section 10.5 of the PBAPS M+SAR (Reference 15) addresses a plant-specific PRA for MELLLA+ conditions in accordance with Limitation and Condition 12.21. As stated in Section 10.10 of this TSAR, “The PBAPS IPE PRA model and analysis will not be specifically updated for TPO because the change in plant risk from the TPO uprate is insignificant.

12. As discussed in Section 10.6 of this TSAR, the sensitivity to the TPO uprate regarding IASCC is small and various programs are currently implemented to monitor the aging of plant components. Thus, the conclusions of the PBAPS M+SAR (Reference 15), Section 10.7.1, Irradiated Assisted Stress Corrosion Cracking, remain valid.

13. As stated in the PBAPS M+SAR (Reference 15), Section 2.6.2, [[

]]

Appendix C - Limitations from Safety Evaluation for LTR NEDC-33075P

Disposition of additional limitations and conditions related to the SE for NEDC-33075P, Revision 7, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density"

There are 4 limitations and conditions listed in Section 5 of the DSS-CD LTR Revision 7 SER. The table below lists each of the 4 limitations and conditions and identifies which section of the M+SAR discusses compliance with each limitation and condition.

Limitation and Condition Number from NRC SER	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
5.1	The NRC staff previously reviewed and approved the implementation of DSS-CD using the approved GEH Option III hardware and software. The DSS-CD solution is not approved for use with non-GEH hardware. The hardware components required to implement DSS-CD are expected to be those currently used for the approved Option III. If the DSS-CD hardware implementation deviates from the approved Option III solution, a hardware review by the NRC staff will be required. Implementations on other Option III platforms will require plant-specific reviews.	Comply	(1)
5.2	The CDA setpoint calculation formula and the adjustable parameters values are defined in NEDC-33075P, Revision 7 (Reference 8). Deviation from the stated values or calculation formulas is not allowed without NRC review. To this end, the subject TR, when approved and implemented by a licensed nuclear power plant, must be referenced in the plant TSs, so that these values become controlled and part of the licensing bases.	Comply	(2)
5.3	The NRC staff previously concluded that the plant-specific settings for eight of the FIXED parameters and three of the ADJUSTABLE parameters, as stated in section 3.6.3 of the NRC staff's SE for NEDC-33075P, Revision 5 (Reference 57), are licensing basis values. The process by which these values will be controlled must be addressed by licensees.	Comply	(3)

Limitation and Condition Number from NRC SER	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
5.4	If plants other than Brunswick Steam Electric Plant, Units 1 and 2, use the DSS-CD trip function, those plant licensees must ensure the DSS-CD trip function is applicable in their plant licensing bases, including the optional BSP trip function, if it is to be installed.	Comply	(4)

Notes:

1. As discussed in the PBAPS M+SAR (Reference 15), Section 2.4, the DSS-CD solution is implemented on GEH hardware that is currently installed and approved by the NRC for the Option III solution.
2. As discussed in the PBAPS M+SAR (Reference 15), Section 2.4.1, the subject topical report, or GESTAR II, is incorporated into the PBAPS Units 2 and 3 TSs.
3. The values of the FIXED and ADJUSTABLE parameters are established by GEH and will be documented in a DSS-CD Settings Report.
4. Verification and validation of the DSS-CD trip function code was performed for transportability considerations.

Appendix D - Limitations and Conditions Applicable to the Use of TRACG04 / PANAC11 in ATWS Overpressure Analyses

This appendix addresses limitations and conditions relevant to the use of TRACG04 / PANAC11 in the analysis of the ATWS overpressure event. The use of TRACG04 / PANAC11 in this application has been previously reviewed by the NRC, as outlined below:

- NEDE-32906P-A, Revision 3 (Reference 58) represents approval of the application of TRACG for AOOs. There are five limitations and conditions for AOOs in Revision 3 that were included in the original Revision 0 SE dated October 21, 2001. The approval of Supplement 1 by the SE dated August 18, 2003 effectively negates Limitation and Condition 3 of the original Revision 0 approval.
- NEDE-32906P, Supplement 1-A (Reference 50) represents the approval of TRACG02 / PANAC10 for ATWS overpressure events. There are four limitations and conditions within Supplement 1-A and an additional statement of information required when a licensee uses TRACG for ATWS analysis in a license amendment.
- NEDC-32906P, Supplement 3-A, Revision 1 (Reference 13) represents the migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for AOO transients and ATWS overpressure events. The only Supplement 3-A limitation and condition required for submittal of this LAR is Limitation and Condition 4.33, which is dispositioned in this appendix.

The four limitations and conditions from the NRC SE of NEDE-32906P, Supplement 1-A are addressed in Table D-1 below. Supplement 3-A Limitation and Condition 4.33, Submittal Requirements Condition, replicates statements from Section 4, Conditions and Limitations, of Supplement 1-A and is addressed in Table D-2 below. Table D-3 in this appendix supplies the chosen parameters and conservative nature of the input parameters, as required in Supplement 3-A Limitation and Condition 4.33, Item 2.

Table D-1 NEDE-32906P, Supplement 1-A Limitations and Conditions

Limitation and Condition Number from NRC SER	Limitation and Condition Description	Disposition	Section of PBAPS TSAR which addresses the Limitation and Condition
4.1	Application of the methodology is considered for prediction of the reactor vessel peak pressure only. The prediction is to be terminated at the time of the signal to initiate SLCS pump injection of boron into the reactor coolant system.	Comply	(1)
4.2	Simply referring to MELLLA+ in the application of the TRACG methodology to ATWS is not sufficient. The flow rate and power level used in the individual applications must be clearly stated and the power-to-flow ratios must not be outside the ranges used in this review. MELLLA+ is not applicable to the BWR/2 class of plants. This point needs to be made in the approved version of the LTR.	Comply	Section 9.3.1
4.3	For each application of the TRACG ATWS methodology, it must be made clear exactly what power level is being used, not only the percentage of licensed power, but the actual power level.	Comply	Section 9.3.1
4.4	Application of TRACG to ATWS events assumes there is no thermal/hydraulic -neutronic instability. The methodology has not been reviewed for applicability to instability conditions.	Comply	(2)

Notes:

1. The TRACG ATWS overpressure analysis simulates the event for 30 seconds, which is sufficient for demonstrating the peak vessel pressure.
2. The TRACG ATWS Overpressure LTR is not applicable to the TRACG ATWSI analysis. A plant-specific TRACG ATWSI analysis is performed to demonstrate compliance with the PCT acceptance criteria.

Table D-2 NEDE-32906P Supplement 3-A, Limitation and Condition 4.33

From NEDE-32906P Supplement 3-A, Limitation and Condition 4.33, Submittal Requirements Condition, "The NRC staff also notes that a generic LTR describing a code such as TRACG cannot provide full justification for each specific individual plant application. When a licensee proposes to reference the TRACG-based ATWS methodology for use in a license amendment, the individual licensee or applicant must provide justification for the specific application of the code in its request which is expected to include:"

Additional Justification Requirement Number from NRC SER	TRACG LTR Applicability Requirement	Applicability Assessment Parameter	Disposition
1	Nodalization	Specific guidelines used to develop the plant-specific nodalization. Deviations from the reference plant must be described and defended.	The nodalization is consistent with the TRACG AOO LTR with increased axial nodes in the bypass region consistent with the TRACG stability nodalization procedures.
2	Chosen Parameters and Conservative Nature of Input Parameters	A table that contains the plant-specific parameters and the range of the values considered for the selected parameter during the approval process. When plant-specific parameters are outside the range used in demonstrating acceptable code performance, the licensee or applicant will submit sensitivity studies to show the effect of that deviation.	The plant-specific operating parameters noted in Table D-3 below are compared to the parameters from Table 8-4 of NEDE-32906P Supplement 1-A.
3	Calculated Results	The licensee or applicant using the approved methodology must submit the results of the plant-specific analyses reactor vessel peak pressure.	The TRACG vessel peak pressure results are provided in TSAR Section 9.3.1.

Table D-3 Plant-Specific Applicability Comparison for TRACG ATWS Overpressure LTR (NEDE-32906P, Supplement 1-A) Parameters

Parameter	TRACG ATWS Overpressure LTR Parameter Range	Plant-Specific Parameter Value	Disposition
Total Core Power	The core power range of 90% to 100% power is evaluated. The basis chosen for plant specific analysis is 100% power.	The PBAPS TPO ATWS analysis is evaluated at 101.7% CLTP consistent with the limiting core power basis provided in the TRACG ATWS Overpressure LTR.	Comply
Total Core Flow	The CF range of 73% to 100% rated core flow (RCF) is evaluated. The basis chosen for plant specific analysis is the minimum CF at RCF.	The PBAPS TPO ATWS analysis is evaluated at 85.2% RCF consistent with the limiting CF basis provided in the TRACG ATWS Overpressure LTR.	Comply
Power to Flow Ratio	The maximum power to flow ratio range is not explicitly defined in the LTR. However, the LTR is applicable to BWR/2 thru BWR/6, which at low CF conditions may achieve a power to flow ratio of 50.0 MWt/Mlbm/hr.	The PBAPS TPO ATWS analysis is evaluated with a power-to-flow ratio of 46.0 MWt/Mlbm/hr.	Comply
Feedwater Temperature	The FW temperature range of rated to a FW temperature reduction of 80°F is evaluated. The basis chosen for plant specific analysis is rated FW temperature. Note, the 80°F temperature reduction is not a PBAPS specific range. The conclusion of the TRACG ATWS Overpressure LTR is applicable to plants with larger FW temperature reductions.	The PBAPS TPO ATWS analysis is evaluated at rated FW temperature consistent with the limiting CF basis provided in the TRACG ATWS Overpressure LTR.	Comply
Steam Dome Pressure	The steam dome pressure range of ± 18 psi from the nominal operating dome pressure is evaluated. The steam dome pressure has been characterized as insensitive.	The PBAPS TPO ATWS analysis is evaluated from the nominal dome pressure.	Comply

Parameter	TRACG ATWS Overpressure LTR Parameter Range	Plant-Specific Parameter Value	Disposition
Downcomer Water Level	The downcomer water level range of ± 12 inches from the nominal operating narrow range level is evaluated. The downcomer water level has been characterized as insensitive.	The PBAPS TPO ATWS analysis evaluates the nominal operating narrow range water level.	Comply
Core Exposure Distribution	The core exposure distribution range of BOC, MOC and EOC is evaluated. The basis chosen for plant specific analysis is BOC exposure.	The PBAPS TPO ATWS analysis is evaluated at BOC exposure consistent with the limiting exposure basis provided in the TRACG ATWS Overpressure LTR.	Comply
Axial Power Distribution	The axial power distribution analysis evaluated in the LTR concluded the parameter to be not sensitive to the initial condition.	The PBAPS TPO ATWS analysis is evaluated with a bottom peak axial power shape.	Comply
MSIV Closure Time	The MSIV closure time range of 3 to 5 seconds is evaluated. The MSIV closure time has been characterized as insensitive.	The PBAPS TPO ATWS analysis evaluates the nominal MSIV closure time of 4 seconds.	Comply
Low Steamline Pressure Isolation Setpoint	The low steamline pressure isolation setpoint range of time 0 (setpoint equal to initial steamline pressure) to the lower AL is evaluated. The earlier (time 0) steamline isolation is limiting for PRFO event. Therefore, the MSIVC event is the limiting event for overpressure.	The PBAPS TPO ATWS analysis evaluates the low steamline pressure isolation setpoint of 825 psig, which is the TS AV.	Comply
SRV and SSV Capacity	The SRV and SSV range of ASME certified capacity to ASME certified capacity + 9.5% is evaluated. The SRV and SSV capacity is limiting at the ASME certified capacity value.	The PBAPS TPO ATWS analysis evaluates the ASME certified capacity.	Comply

ATTACHMENT 9

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

Cameron Affidavit Supporting Withholding Attachment 8 from Public Disclosure

Caldon Ultrasonics Technology Center

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CAMERON
A Schlumberger Company

February 13, 2017
CAW 17-02

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-464 Rev. 4 "Uncertainty Analysis for Thermal Power Determination at Peach Bottom Unit 2 Using the LEFM ✔ + System"

Gentlemen:

This application for withholding is submitted by Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 17-02 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 17-02 and should be addressed to the undersigned.

Very truly yours,

Ernest M. Hauser
Director of Business Development
Nuclear and Defense Markets

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

February 13, 2017
CAW 17-02

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Ernest M. Hauser, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron Holding Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Ernest M. Hauser
Director of Business Development
Nuclear and Defense Markets

Sworn to and subscribed before me

this 13th day of

February, 2017

Frances A. Lewis
Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Frances A. Lewis, Notary Public
Coraopolis Boro, Allegheny County
My Commission Expires Nov. 25, 2018
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

1. I am the Director of Business Development for Nuclear and Defense Markets of Caldon Ultrasonics Technology Center, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

5. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Cameron.

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:
Cameron Engineering Report ER- 464 Rev. 4 "Uncertainty Analysis for Thermal Power Determination at Peach Bottom Unit 2 Using the LEFM ✔ + System"

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Peach Bottom Atomic Power Station for flow measurement at the licensed reactor thermal power level of 4016 MWt.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

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CAMERON
A Schlumberger Company

February 13, 2017
CAW 17-01

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-463 Rev. 4 "Uncertainty Analysis For Thermal Power Determination at Peach Bottom Unit 3 Using the LEFM ✔ + System"

Gentlemen:

This application for withholding is submitted by Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 17-01 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 17-01 and should be addressed to the undersigned.

Very truly yours,

Ernest M. Hauser
Director of Business Development
Nuclear and Defense Markets

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

February 13, 2017
CAW 17-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Ernest M. Hauser, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Ernest M. Hauser
Director of Business Development
Nuclear and Defense Markets

Sworn to and subscribed before me

this 13th day of

February, 2017

Frances A. Lewis
Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Frances A. Lewis, Notary Public
Coraopolis Boro, Allegheny County
My Commission Expires Nov. 25, 2018
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

1. I am the Director of Business Development for Nuclear and Defense Markets of Caldon Ultrasonics Technology Center, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

5. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Cameron.

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Engineering Report ER-463 Rev. 4 "Uncertainty Analysis for Thermal Power Determination at Peach Bottom Unit 3 Using the LEFM ✓+ System"

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Peach Bottom Atomic Power Station for flow measurement at the licensed reactor thermal power level of 4016 MWt.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

ATTACHMENT 11

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

WEC Affidavit Supporting Withholding Attachment 10 from Public Disclosure

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
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Rockville, MD 20852

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CAW-16-4497

October 27, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-BWR-ENG-16-032-P, Revision 0, "Peach Bottom Units 2 & 3 Steam Dryer Report at MUR Conditions" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4497 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Exelon Generation.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4497, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

J. A. Gresham

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

JAGresham

James A. Gresham, Manager
Regulatory Compliance

Date: 10/27/16

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-BWR-ENG-16-032-P, Revision 0, "Peach Bottom Units 2 & 3 Steam Dryer Report at MUR Conditions" (Proprietary), for submittal to the Commission, being transmitted by Exelon Generation letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the high-cycle fatigue assessment of the steam dryers at Peach Bottom Atomic Power Station (PBAPS) Units 2 & 3 at Measurement Uncertainty Recovery (MUR) conditions, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to assist Exelon Generation in fulfilling the requirements specified in the PBAPS Units 2 & 3 Renewed Facility Operating Licenses.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of plant specific steam dryer analysis for licensing basis applications.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC associated with the high-cycle fatigue assessment of the steam dryers at Measurement Uncertainty Recovery (MUR) conditions for PBAPS Units 2 & 3 with regards to the PBAPS Units 2 & 3 license conditions, and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary that Westinghouse customarily holds in confidence is identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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ATTACHMENT 12

Peach Bottom Atomic Power Station Units 2 and 3

Renewed Facility Operating License Nos. 50-277 and 50-278

Westinghouse Electric Company, Peach Bottom Units 2 and 3 Steam Dryer Report at MUR Conditions, LTR-BWR-ENG-16-032-NP, Revision (Non-Proprietary Version)

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066
USA

LTR-BWR-ENG-16-032-NP

Peach Bottom Units 2 and 3
Steam Dryer Report at MUR Conditions

Revision 0

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1 INTRODUCTION AND SUMMARY

Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3 are currently operating at full extended power uprate (EPU) conditions and plan to implement Measurement Uncertainty Recovery (MUR) which would result in an increase of approximately 1.7%. A high-cycle fatigue assessment of the PBAPS Units 2 and 3 replacement steam dryers (RSDs) has been completed utilizing main steam line (MSL) data extrapolated to MUR conditions. The assessment is a complete reanalysis using the Westinghouse steam dryer acoustic/structural methodology, Acoustic Circuit Enhanced (ACE) Revision 3.1. ACE Revision 3.1 includes the end-to-end biases and uncertainties (B/U) from the PBAPS Unit 2 benchmarking at EPU conditions. Additionally, the reanalysis considers the effects of non-main steam line acoustic (NMSLA) loads. This report provides the results of the high-cycle fatigue assessments for PBAPS Units 2 and 3 steam dryers. The effects of Maximum Extended Load Line Limit Analysis Plus (MELLLA+) conditions have also been assessed.

An assessment was also performed to show compliance of the Peach Bottom Units 2 and 3 steam dryers with the structural requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NG. The assessment shows that the dryers [

] a,b,c

Based on the assessments performed, it has been determined that the minimum alternating stress ratio (MASR) at predicted MUR conditions (4016 MWt) including MELLLA+ for PBAPS Unit 2 [ ] a,b,c For PBAPS Unit 3 [ ] a,b,c Therefore, all steam dryer locations have a stress ratio greater than the acceptance limit of 1.0.

2 METHODOLOGY

The PBAPS Units 2 and 3 steam dryers were analyzed using the same methods used to structurally qualify the steam dryers for Extended Power Uprate (EPU) operating conditions.

2.1 ACE Revision 3.1 Acoustic Analysis

[

] a,b,c

Table 2-1 Measured Natural Frequencies a,b,c

[

]

[

] a,b,c

Table 2-2 Peach Bottom Power Levels a,b,c

[ ]

[

] a,b,c

Figures 2-1 through Figure 2-8 in Section 2.1.1 and Section 2.1.2 show [

]a,b,c

The bump-up factors for each MSL are shown on each figure and are summarized in Table 2-3.

Table 2-3 Bump-Up Factors a,b,c

[ ]

2.1.1 Bump-up Factors Unit 2

a,b,c

Figure 2-1 Unit 2 RMS Pressure [ ] a,b,c

a,b,c

Figure 2-2 Unit 2 RMS Pressure [ ] a,b,c

a,b,c

Figure 2-3 Unit 2 RMS Pressure [ ] a,b,c

a,b,c

[ ]

Figure 2-4 Unit 2 RMS Pressure [ ] a,b,c

2.1.2 Bump-up Factors Unit 3

a,b,c

Figure 2-5 Unit 3 RMS Pressure [ ] [a,b,c]

a,b,c

Figure 2-6 Unit 3 RMS Pressure [ ] a,b,c

a,b,c

[ ]

Figure 2-7 Unit 3 RMS Pressure [ ] a,b,c

a,b,c

Figure 2-8 Unit 3 RMS Pressure [ ] a,b,c

2.2 Non-MSL Acoustic Analysis

The NMSLA analysis methodology used to evaluate the steam dryer at EPU operating conditions (Reference 2) is also used to evaluate the steam dryer at MUR operating conditions. [

] a,b,c

Per Reference 2, comparative analyses were performed using the [

] a,b,c

3 HIGH CYCLE FATIGUE AND ASME ANALYSES SUMMARY

A structural evaluation was performed considering both the [ ] a,b,c This evaluation was performed for Peach Bottom Unit 2 without the instrumentation mast and for Peach Bottom Unit 3. The evaluation was performed at predicted MUR and MELLLA+ conditions. [ ] a,b,c

[ ] a,b,c

Table 3-1 summarizes the limiting steam dryer high cycle fatigue stress ratios for Unit 2 and Table 3-2 summarizes the limiting stream dryer high cycle fatigue stress ratios for Unit 3. [ ] a,b,c

Table 3-1 PBAPS U2 Stress Ratio Summary at MUR Conditions

a,b,c

Table 3-2 PBAPS U3 Stress Ratio Summary at MUR Conditions a,b,c

[ ]

An ASME assessment was performed to show compliance of the Peach Bottom Units 2 and 3 steam dryers with the structural requirements of ASME B&PV Code, Section III, Subsection NG. [

] a,b,c

4 CONCLUSIONS

The results of the high cycle fatigue assessment verify the continued structural integrity of the PBAPS Units 2 and 3 steam dryers at MUR conditions. The evaluation of the Units 2 and 3 steam dryers included a complete reanalysis using ACE Revision 3.1, with end-to-end bias and uncertainties from Peach Bottom Unit 2 benchmarking at EPU conditions.

Based on the assessments performed, it has been determined that the MASR at predicted MUR conditions including MELLLA+ (4016 MWt) for PBAPS Unit 2 [

][a,b,c] Therefore, all steam dryer locations have a stress ratio greater than the acceptance limit of 1.0.

5 REFERENCES

5. Exelon Letter to NRC, "Extended Power Uprate: Results of Unit 2 Replacement Steam Dryer Power Ascension Testing," dated August 7, 2015, Attachment 1, Westinghouse Proprietary Document, "Transmittal of Peach Bottom Unit 2 Replacement Steam Dryer Report at EPU Conditions," LTR-BWR-ENG-15-066-P, August 6, 2015 (ADAMS Accession No. ML15219A618).

6. Exelon Letter to NRC, "Extended Power Uprate: Unit 3 Replacement Steam Dryer Revised Analysis Report, Supplement 1," dated October 2, 2015, Attachment 1, Westinghouse Proprietary Document, "Transmittal of Peach Bottom Unit 3 Replacement Steam Dryer Report at Predicted EPU Conditions," LTR-BWR-ENG-15-076-P, Rev. 1, October 2, 2015 (ADAMS Accession No. ML15275A168).

7. Exelon Letter to NRC, "License Amendment Request – Extended Power Uprate," dated September 28, 2012, Attachment 17.B.5, Westinghouse Proprietary Report, "Peach Bottom Unit 2 and Unit 3 Replacement Steam Dryer Four-Line Subscale Acoustic Test Data Evaluation and Derivation of CLTP-to-EPU Scaling Spectra," WCAP-17611-P Revision 1, August 2012 (ADAMS Accession No. ML12286A012).

8. Exelon Letter to NRC, "MELLLA+ License Amendment Request – Supplement 4 Response to Request for Additional Information," dated July 6, 2015. (Attached to this letter in EDMS)