~~Withhold under 10 CFR 2.390. Enclosures 7, 10, and 11 contain Proprietary Information. Enclosure 15 contains Security-Related Information~~

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10 CFR 50.90

June 28, 2016
GO2-16-096

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSE AMENDMENT REQUEST TO REVISE OPERATING LICENSE AND TECHNICAL SPECIFICATIONS FOR MEASUREMENT UNCERTAINTY RECAPTURE (MUR) POWER UPRATE**

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

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, "Application for Amendment of License or Construction Permit," and 10 CFR 50, Appendix K, "ECCS Evaluation Models," Energy Northwest hereby requests a license amendment to revise the Columbia Generating Station (Columbia) Renewed Facility Operating License (OL) NPF-21 and Technical Specifications (TS). Specifically, the proposed changes revise the OL and TS to implement an increase in rated thermal power from the current licensed thermal power (CLTP) of 3486 megawatts thermal (MWt) to 3544 MWt.

The proposed changes are based on reduced uncertainty in the feedwater flow and temperature measurement that reduces the total power level measurement uncertainty, which is achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The LEFM instrumentation was installed at Columbia during the spring 2015 refueling outage.

The content of this request is in accordance with the guidance contained in NRC RIS 2002-03. Energy Northwest has only proposed those OL and TS changes that are required to implement the increased power level. Additionally, Energy Northwest has reviewed the requests for additional information (RAI) from the facilities identified in Enclosure 1, Section 4.2, "Precedents," and has included information within the body of the submittal to address the general topics of those requests.

This submittal contains the following Enclosures:

Enclosure 1	Description and Evaluation of the Proposed Change
Enclosure 2	Markup of Existing Renewed Facility Operating License and Technical Specifications
Enclosure 3	New Licensee Controlled Specification and Markup of Technical Specification Bases "For Information Only"
Enclosure 4	Revised (Clean) Renewed Facility Operating License and Technical Specification Pages
Enclosure 5	Regulatory Issue Summary (RIS) 2002-03 Cross Reference
Enclosure 6	Summary of Regulatory Commitments
Enclosure 7	General Electric-Hitachi (GEH) Report NEDC-33853P, "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization," Revision 0 (Proprietary Version)
Enclosure 8	Affidavits from GEH and the Electric Power Research Institute (EPRI) Supporting the Withholding of Information in Enclosure 7 from Public Disclosure
Enclosure 9	GEH Nuclear Report NEDO-33853, "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization," Revision 0 (Non-Proprietary Version)
Enclosure 10	Cameron (Caldon) Document ER-1049, "Bounding Uncertainty Analysis for Thermal Power Determination at Columbia Nuclear Generating Station Using the LEFM CheckPlus System," Revision 3 (Proprietary Version)
Enclosure 11	Cameron (Caldon) Document ER-1074, "Meter Factor Calculation and Accuracy Assessment for Columbia Nuclear Generating Station," Revision 0 (Proprietary Version)
Enclosure 12	Affidavits from Cameron International Corporation Supporting the Withholding of Information in Enclosures 10 and 11 from Public Disclosure
Enclosure 13	Columbia Calculation NE-02-15-08, "Heat Balance Determination for Rated Thermal Power," Revision 0
Enclosure 14	LEFM Flowmeter Installation Drawings
Enclosure 15	Bonneville Power Administration Report, "Columbia Generating Station Measurement Uncertainty Recapture Reactor Thermal Power Limit Uprate Study," dated June 22, 2016 (Security-Related Information)

Approval of the proposed amendment is requested by May 13, 2017, prior to the start of the 2017 Refueling Outage (RFO). If approved prior to the 2017 RFO, the instrument recalibrations required for implementation will occur during the 2017 RFO. If not, the amendment will be implemented within 120 days of approval.

In accordance with 10 CFR 50.91, "Notice for Public Comment; State Consultation," Energy Northwest is notifying the State of Washington of this amendment request by transmitting a copy of this letter and enclosures to the designated State Official.

In accordance with 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding," Columbia requests withholding from public disclosure Enclosures 7, 10, 11, and 15. Enclosure 7 contains information that is considered proprietary by GEH Nuclear Energy and the Electric Power Research Institute (EPRI). Affidavits supporting this request are provided in Enclosure 8 and a non-proprietary version of Enclosure 7 is provided in Enclosure 9. Enclosures 10 and 11 are considered proprietary by Cameron International Corporation. Affidavits supporting these requests are included in Enclosure 12. Non-proprietary versions of Enclosures 10 and 11 are not available. Enclosure 15 to this letter provides the grid study and contains information deemed by the Bonneville Power Administration to be security sensitive information related to critical infrastructure. Energy Northwest requests that Enclosure 15 be withheld from public disclosure in accordance with 10 CFR 2.390(d)(1). Upon removal of Enclosures 7, 10, 11, and 15, this letter is decontrolled.

Regulatory commitments associated with this submittal are identified in Enclosure 6.

If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

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

Executed on this 27th day of June, 2016.

Respectfully,

A. L. Javorik

Vice President, Engineering

Enclosures: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda – BPA 1399 (email)
WA Horin – Winston & Strawn (email)
RR Cowley – WDOH (email)
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License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 1

Description and Evaluation of the Proposed Change

Description and Evaluation of the Proposed Change

Contents

1.0 Summary Description

Pursuant to 10 CFR 50.90, “Application for Amendment of License or Construction Permit,” and 10 CFR 50, Appendix K, “ECCS Evaluation Models,” Energy Northwest hereby requests a license amendment to revise the Columbia Generating Station (Columbia) Renewed Facility Operating License (OL) No. NPF-21 and Technical Specifications (TS). Specifically, the proposed changes revise the OL and TS to implement an increase in rated thermal power (RTP) from the current licensed thermal power (CLTP) of 3486 megawatts thermal (MWt) to a measurement uncertainty recapture (MUR) thermal power of 3544 MWt. Columbia was originally licensed to 3323 MWt, and in 1995 a power uprate amendment authorized an increase in power to the CLTP of 3486 MWt.

The proposed changes are based on reduced uncertainty in the feedwater flow and temperature measurement that reduces the total power level measurement uncertainty. This is achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The LEFM system was installed at Columbia during the spring 2015 refueling outage (RFO).

2.0 Detailed Description

The proposed changes to the OL and TS are described below, with marked-up pages included in Enclosure 2.

A proposed new section to be added to the Licensee Controlled Specifications (LCS) and proposed changes to the TS Bases are also described below, with marked-up pages included in Enclosure 3. These changes are for information only, and do not require NRC approval.

2.1 Columbia Renewed Facility Operating License (OL)

Changes related to the value of RTP for Columbia, OL No. NPF-21, Section 2.C.(1), “Maximum Power Level,”

Current: The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3486 megawatts thermal).

Proposed: The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3544 megawatts thermal).

2.2 Columbia TS 1.1, Definition of Rated Thermal Power (RTP)

Current: RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.

Proposed: RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3544 MWt.

2.3 Columbia TS 3.3.1.1, "RPS Instrumentation *(After Implementation of PRNM Upgrade)*"

- REQUIRED ACTION E.1

  Current: Reduce THERMAL POWER to < 30% RTP

  Proposed: Reduce THERMAL POWER to < 29.5% RTP

- Surveillance Requirement (SR) 3.3.1.1.12

  Current: Verify Turbine Throttle Valve - Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.

  Proposed: Verify Turbine Throttle Valve - Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 29.5% RTP.

- TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b "Average Power Range Monitors Simulated Thermal Power - High", Allowable Value

  Current value: ≤ 0.63W + 64.0% RTP and ≤ 114.9% RTP[(c)]

  Proposed value: ≤ 0.62W + 62.9% RTP and ≤ 114.9% RTP[(c)]

- Table 3.3.1.1-1 Note (c)

  Current: ≤ 0.63W + 60.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

  Proposed: ≤ 0.62W + 59.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

- Table 3.3.1.1-1 Function 8, "Turbine Throttle Valve - Closure", Applicable Modes or Other Specified Conditions

  Current value: ≥ 30% RTP

  Proposed value: ≥ 29.5% RTP

- Table 3.3.1.1-1 Function 9, "Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low", Applicable Modes or Other Specified Conditions

  Current value: ≥ 30% RTP

  Proposed value: ≥ 29.5% RTP

2.4 Columbia TS 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation"

- APPLICABILITY

  Current: THERMAL POWER ≥ 30% RTP

  Proposed: THERMAL POWER ≥ 29.5% RTP

- REQUIRED ACTION C.2.

  Current: Reduce THERMAL POWER to < 30% RTP

  Proposed: Reduce THERMAL POWER to < 29.5% RTP

- SR 3.3.4.1.3

  Current: Verify TTV – Closure and TGV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP

  Proposed: Verify TTV – Closure and TGV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 29.5% RTP

2.5 Columbia TS 3.3.6.1 “Primary Containment Isolation Instrumentation”

- Table 3.3.6.1-1, “Primary Containment Isolation Instrumentation”, Function 1.c, “Main Steam Line Flow - High”, Allowable Value

  Current value: ≤ 124.4 psid

  Proposed value: ≤ 137.9 psid

2.6 Columbia Licensee Controlled Specifications (LCS) Changes, New Section (Information Only)

New LCS Section 1.3.9, “LEFM Feedwater Flow Instrumentation,” is added to specify the proposed requirements and bases for the LEFM system and to specify a surveillance requirement.

2.7 TS Bases Changes (Information Only)

- The Bases for Sections 3.3.1.1 and 3.3.4.1 are changed to provide supporting bases discussions for the required TS changes identified in Section 2.3 through 2.5 above.
- The Bases for Section 3.3.2.2, “Feedwater and Main Turbine High Water Level Trip Instrumentation,” are changed to incorporate the RTP value above which the Level 8 trip indirectly initiates a reactor scram from the main turbine trip.
- The Bases for Section 3.7.6, “Main Turbine Bypass System,” are changed to reflect the bypass capacity of the system based on the revised steam flow of the main steam system.

Details of the aforementioned changes are provided in Enclosure 3.

3.0 Technical Evaluation

3.1 Background and 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 was made to this paragraph, which

became effective on July 31, 2000, that allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error. Utilization of the Cameron LEFM system at Columbia has resulted in reduced uncertainty in feedwater flow and temperature measurement that reduces the total power level measurement uncertainty. The core thermal power measurement uncertainty is described in Section 3.2.3 of this enclosure.

During the 2015 refueling outage, Columbia completed the installation of the LEFM CheckPlus system which included changes to the Plant Process Computer (PPC), the Transient Data Acquisition System (TDAS), the Plant Data Information System (PDIS) and the MONICORE core monitoring system. The LEFM CheckPlus system provides a more accurate reactor feedwater mass flow measurement. The LEFM system measures feedwater flow using ultrasonic pulses, which are digitally processed. Since installation, the LEFM system provides a more accurate feedwater flow input to the thermal heat balance calculation performed by the PPC. This calculated thermal power is used by the control room operators to monitor compliance with the OL condition for CLTP maximum power level, to determine the margins to the power distribution limits (PDL) and to calibrate the average power range monitor (APRM) neutron flux indication to represent actual reactor power. This amendment request, once approved, authorizes the changes identified in Sections 2.1 - 2.5 of this enclosure allowing an increase in RTP to the MUR thermal power of 3544 MWt.

The PPC provides indication and alerts related to the LEFM system. As discussed in the following sections of this enclosure, the PPC is also used to determine the difference between the feedwater flow indication from the LEFM system and the existing reactor feedwater flow venturi instrumentation for the purpose of data validation.

The scope and content of the evaluations performed and described in this request are in accordance with the guidance contained in NRC Regulatory Issue Summary (RIS) 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” (Reference 6.2). Enclosure 5 of this request provides a cross-reference between the contents of this application and the guidance in RIS 2002-03.

The ECCS evaluation and other plant safety analyses currently assume an uncertainty of 2% of the CLTP (3486 MWt). Energy Northwest has evaluated the effects of the proposed increase in RTP using an approach developed by General Electric-Hitachi (GEH) Nuclear Energy and approved by the NRC, which is documented in NEDC-32938P-A, “Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” (Reference 6.6). Enclosure 7 summarizes the results of all significant safety evaluations performed that justify increasing the licensed thermal power. Review of these analyses support the requested license power level increase to 3544 MWt.

3.2 LEFM Measurement and Core Thermal Power Uncertainty

3.2.1 LEFM Feedwater Flow and Temperature Measurement

The ultrasonic feedwater flowmeter installed at Columbia is a Cameron LEFM CheckPlus ultrasonic multi-path, transit time flowmeter. This LEFM system will be used in lieu of the current venturi-based feedwater flow indication and resistance temperature detector (RTD) temperature indication to provide feedwater flow input for the plant thermal heat balance calculation. The currently installed feedwater flow venturis will be used if the LEFM is not functional. The LEFM system uses ultrasonic transit time principles to determine fluid velocity and sound velocity. This flow measurement method is described in Caldon topical reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," Revision 0 (Reference 6.7), and ER-157P, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," Revision 8 and Revision 8 Errata (Reference 6.8). These topical reports were approved by the NRC in documents titled, "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,'" (Reference 6.9) and "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 a Power Uprate with the LEFM Check or CheckPlus System,'" (Reference 6.10).

In References 6.9 and 6.10, the NRC established criteria for use of these topical reports in requests for license amendments. Energy Northwest's response to those criteria is provided in Section 3.2.4 of this enclosure.

Enclosure 10 provides the analysis of the uncertainty contribution of the LEFM CheckPlus system operating in the Check Plus (normal) mode, as well as when operating in the Check (maintenance) mode, to the overall calculated thermal power uncertainty. This analysis is a bounding analysis for Columbia and was completed following the calibration of the LEFM spool pieces. Additionally, the as-built dimensions were inputs for all computations, and confirmed that the uncertainties in these dimensions lie within the bounding values used in the bounding analysis. The commissioning tests for the Columbia LEFM CheckPlus system confirmed that the time measurement uncertainties are within the bounding values used in the analysis.

The LEFM instrumentation is not safety-related. Components such as the spool pieces, system control cabinet and components, pressure transmitters, RTDs, and the power supplies are Quality Class 2, Seismic Category II. The LEFM system was designed and manufactured in accordance with Cameron's Quality Assurance Program. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM system are provided in Reference 6.7, Section 6.4 and Table 6.1.

The LEFM CheckPlus system consists of a measurement spool piece meter in each feedwater line, two transmitter signal processing units per spool piece and two redundant central processing units (CPU). Each measurement spool piece contains 16 ultrasonic, multi-path, transit time transducers grouped into two planes of eight transducers each, two 4-wire RTDs, and two pressure transmitters.

The LEFM system installed at Columbia performs automatic continuous self-checking of the transducer signals and the calculation results. This testing provides verification that the digital circuits are operating correctly and the LEFM system is within its specified accuracy envelope.

The LEFM system has two operating modes as well as a fail mode. Normal operation for the LEFM system is the Check Plus mode. In this mode, both planes of transducers are in service and system operations are processed by both redundant CPUs. If the system is subjected to a failure involving a transducer or failure of one plane of operation due to a transmitter signal processing unit malfunction, the system reverts to the Check mode. The control room operators are provided a visual alarm on the PPC when the LEFM system shifts from the Check Plus mode (normal mode) to the Check mode (maintenance mode).

- Check Plus Mode (normal mode):

  When in the Check Plus mode, a system normal is displayed when all the feedwater flow, temperature, and header pressure signals for feedwater lines A and B are normal and operating within design limits. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is less than 0.3%.

  The plant can operate at ≤ 3544 MWt as discussed in Section 3.2.3 of this enclosure.

- Check Mode (maintenance mode):

  When the LEFM system shifts from the Check Plus mode to the Check mode a visual alarm indicates that there has been a loss of LEFM system redundancy. The LEFM system Check mode indicates a loss of function that causes it to operate outside that specified accuracy envelope of ± 0.3%. Typically, this occurs due to a malfunction of a single path or plane and results in an uncertainty increase to ± 0.5%. In the event of a failure of one path or plane that cannot be restored to full functionality (Check Plus mode) within 72 hours, power will be reduced from 3544 MWt to ≤ 3537 MWt as discussed in Section 3.2.3 of this enclosure. The plant can operate at this power level indefinitely. The operators will be provided with procedural guidance for those occasions when the LEFM system is in the Check mode.

- Fail Mode:

  The LEFM system's Fail mode indicates a loss of function that causes the LEFM system to operate outside the specified accuracy envelope of ± 0.5%. In this case the power level uncertainty reverts to the 2.0% associated with the venturi flow meters and power will be reduced to ≤ 3486 MWt within 72 hours if LEFM functionality cannot be restored.

The LEFM system has continuous operating online self-diagnostic processes to verify that the digital circuits are operating correctly and within the design basis uncertainty limits. These processes can identify failure conditions that will cause the LEFM to switch from the Check Plus mode to the Check mode or to the Fail mode. Validated LEFM data including calculated results, status, and signal process information is sent to the PPC at regular intervals. Calculated LEFM results are compared to venturi data and RTD instrument results as a means of further data validation.

The PPC will provide a visual alarm upon change in the LEFM system status on the operator overview display screen. This includes a change from the Check Plus mode to the Check mode or LEFM Fail mode and requires entry into the Compensatory Measures of the new LCS 1.3.9 as described in Enclosure 3. A visual alarm is provided on the operator overview display screen for sustained loss of data between the LEFM and PPC. In addition to a visual alarm, a loss of data link results in indication that entry into the Compensatory Measures of LCS 1.3.9 is required. Core thermal power calculations automatically revert to calibrated venturi output when the PPC does not have a valid LEFM signal.

When LEFM operation is governed by one of the LCS Conditions, the remaining Completion Time for the Required Compensatory Measures will be displayed (e.g., 71.5 hours remaining until Columbia is derated to 3486 MWt). Maximum allowed core thermal power (CTP) and indications of compliance with the maximum allowed CTP based on LEFM status are displayed.

The 72-hour Completion Time begins when the PPC screens located at the Reactor Operator and Control Room Supervisor stations begin flashing a predetermined warning. The warnings reflect the following conditions:

- System status changes from Check Plus to Check mode due to:
    - one LEFM feedwater flow meter in Check mode and one LEFM feedwater flow meter in the Check Plus mode, or
    - both LEFM feedwater flow meters in Check mode.
- System status change to Fail mode due to one or both LEFM feedwater flow meters in Fail mode.
- Validated loss of signal between LEFM and PPC.

Additionally, there are PPC displays that the operators can use to display detailed information about the LEFM connection status and the function of the LEFM components. This includes detailed information for transducers, the signal processing function and the CPU status. Columbia has two fully redundant PPCs. Each PPC includes redundant processes to collect data from each of the redundant CPU's in the LEFM cabinet.

Methods to determine LEFM system status and the cause of alarms are described in Cameron documentation which will be used to develop specific procedures for operators and maintenance response actions. Justification for the 72-hour completion time is provided in Section 3.2.4 of this enclosure.

3.2.2 Plant Implementation

The Columbia LEFM system was installed and commissioned in accordance with the appropriate Cameron installation and testing procedures. The LEFM measurement spool pieces were installed in the feedwater piping of the two feedwater lines as shown in the installation drawings provided in Enclosure 14.

The installations in feedwater lines A and B are located in straight sections of 24 inch feedwater pipe about 20 feet downstream of the existing feedwater flow venturis. Both spool pieces are located sufficiently remote from major hydraulic disturbances as required by Cameron spool piece installation specifications.

The transducers are located in the turbine building (TB) steam tunnel extension at the TB 501 foot elevation. The integrated gamma dose for 40 years of normal plant operation is 9.5E5 Rads. The material in the LEFM transducers has been exposed to gamma irradiation levels of 10 to 100 Mega Rads with negligible degradation in transducer performance. The system control cabinet is located outside the steam tunnel extension in an area with no significant gamma dose. Therefore, no radiation damage or degradation to the instruments due to the exposure levels in the plant is anticipated.

Following installation, testing included an inservice leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. Final commissioning testing is described in Cameron's LEFM CheckPlus Flow Measurement System Installation and Commissioning Manual for Columbia Nuclear Power Plant (March 2014) (Reference 6.12). All testing was completed satisfactorily in July of 2015.

3.2.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology

Enclosure 10 provides an analysis of the uncertainty contribution of the LEFM CheckPlus system when operating in the Check Plus mode, as well as when operating in the Check mode, to the overall calculated thermal power uncertainty. At Columbia with the LEFM CheckPlus system in the Check Plus mode, calculated core thermal power uncertainty due to the LEFM system is ± 0.276%. In the Check mode, calculated core thermal power uncertainty due to the LEFM system is ± 0.485%. These uncertainties were calculated using the methodology described in Reference 6.8, which was approved by the NRC in Reference 6.10. These uncertainties were rounded up to 0.3% and 0.5% respectively, in the heat balance uncertainty calculation (Enclosure 13).

The measurement uncertainty recapture allows a licensed power level that maintains margin to 102% of CLTP. In Enclosure 13, 102% of 3486 MWt (3556 MWt) was used as a maximum value when determining the MUR power uprate value. The total thermal power heat balance calculation uncertainty is obtained by combining the input uncertainties as random terms except for control rod drive and reactor water cleanup flows which may have dependency due to a PPC bias, thus they are conservatively added together. This results in the following thermal power uncertainties and proposed power levels. The method used in performing the above calculation is based on Energy Northwest Standard EES-4, “Setpoint Methodology “.

- For the LEFM system operating in Check Plus mode, the heat balance calculation has an uncertainty of ±11.649 MWt. This results in a power level of 3556 MWt - 11.649 MWt = 3544.351 MWt. The proposed power level in the Check Plus mode is rounded down to 3544 MWt. Therefore the requested increase in power is approximately 1.66% above the CLTP of 3486 MWt.
- For the LEFM system operating in the Check mode, the heat balance calculation has an uncertainty of ±18.586 MWt. This results in a power level of 3556 MWt - 18.586 MWt = 3537.414 MWt. The proposed power level in the Check (maintenance) mode is rounded down to 3537 MWt.

A revised heat balance calculation has been added to the PPC to support feedwater input from the LEFM system and the existing venturi flow nozzles.

Caldon Topical Report ER-157P, Revision 8 (Reference 6.8), states that the redundancy inherent in the two measurement planes of an LEFM CheckPlus system also makes this system more resistant to total failure when compared to the LEFM Check system. For any single component failure, continued operation at a power greater than that prior to the MUR power uprate can be justified with the LEFM system since the system with the failure is no less than an LEFM Check system.

The NRC SER (Reference 6.10) approving ER-157P, Revision 8 required licensees referencing ER-157P, Revision 8 to ensure compliance with these two limitations/conditions:

1. Continued operation at the pre-failure power 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.
2. The only mechanical difference that potentially affects the Topical Report ER-157P, Revision 8 statement above is that the LEFM CheckPlus system has 16 transducer housing interfaces with the flowing water, whereas the LEFM Check System has 8. Consequently, a LEFM CheckPlus system operating with a single failure that is assumed to disable one plane of transducers is not identical to an LEFM Check system. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate as stated. An acceptable quantification method is to establish the effect in an acceptable test configuration such as can be accomplished at the Alden Laboratory.

Cameron reports ER-1049 (Enclosure 10) and ER-1074 (Enclosure 11) identify the uncertainties associated with LEFM operation in the Check Plus mode and Check mode, including meter factor uncertainties specific to Columbia. These uncertainties were established by the calibration tests performed at Alden Research Laboratory. The impact of a failure disabling one plane of transducers on the LEFM system installed at Columbia has been quantified with an uncertainty of less than ±0.5%. The associated increase in uncertainty from 0.3% to 0.5% results in a maximum allowable power level for this condition of 3537 MWt.

In the event the LEFM system is non-functional (Fail mode), the heat balance calculation will use the existing feedwater venturi flow nozzles until the LEFM system is returned to functional status. To ensure that the venturi-based heat balance calculation is consistent with the LEFM system based heat balance calculation, the venturi-based flow rate will be normalized to the pre-failure LEFM system flow rate.

The loss of the data link between the LEFM system and the PPC (beyond that associated with anticipated data flow interruptions) or a PPC failure will require reducing core thermal power to ≤ 3486 MWt within 72 hours. It is conservative to limit the power within 72 hours to this level until the LEFM system is returned to functional status. A new proposed LEFM feedwater flow instrumentation specification will be added to the LCS, as shown in Enclosure 3, to provide operators with actions to be taken when the LEFM system is not in the normal mode.

This meets the two limitations/conditions identified above.

3.2.4 Disposition of NRC Criteria for Use of LEFM Topical Reports

In References 6.9 and 6.10, the NRC established criteria to be addressed by licensees incorporating the LEFM methodology into the licensing basis. The criteria are listed below, along with a discussion of how each is or 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>

Installation of the LEFMs included development of the necessary procedures and documents required for maintenance and calibration of the LEFM system. Plant maintenance and calibration procedures have been revised to incorporate Cameron's maintenance and calibration requirements. Initial preventive maintenance scope and frequency are based on vendor recommendations. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

For instrumentation other than the LEFM system that contributes to the thermal power heat balance computation, calibration and maintenance is performed periodically using existing site procedures. Instrument channel accuracy, drift, calibration error and instrument error were evaluated and accounted for within the thermal power uncertainty calculation.

The LEFM system software and the PPC software configuration is maintained using existing Columbia procedures, which include verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM system and the instrumentation that contributes to the heat balance calculation is maintained in accordance with Columbia configuration control procedures.

Columbia programs and procedures addressing corrective actions, reporting deficiencies, and receiving and evaluating manufacturer's deficiency reports are discussed in Section 3.2.5, "Deficiencies and Corrective Actions."

<u>LEFM Non-functionality and the Effect on Thermal Power Measurements and Plant Operations</u>

The redundancy inherent in the two measurement planes of an LEFM system as described in Enclosure 10 makes the system tolerant to component failures. Continuously operating online self-diagnostic testing is provided to verify that the digital circuits are operating correctly and within the design basis uncertainty limits. LEFM system malfunctions result in PPC alarm messages to alert the operators if the status of the LEFM instrumentation changes. In these cases, the proposed LCS Compensatory Measures will be applied. Additionally, if the interface between the LEFM system and the PPC has failed, the LEFM will be considered non-operational and the proposed 72 hour allowed outage time would be entered and the LCS Compensatory Measures will be applied. As provided in Enclosure 3, the new LCS Requirements for Operation (RFO) 1.3.9, Feedwater Flow Instrumentation, will be implemented prior to raising thermal power above the CLTP (See Enclosure 6, Item 1).

The proposed LCS specification requires verification that each LEFM system meter is in the Check Plus mode every 24 hours. In addition to this confirmation of status, the PPC alarm messages described above alert the operators if the status of the LEFM instrumentation changes.

The existing feedwater flow venturi-based signals were calibrated using the LEFM system measured feedwater flow at the beginning of operating cycle 23, following the commissioning of the LEFM. The venturi calibration is revalidated and adjusted at the beginning of each cycle when the LEFM is operational at full power conditions. During the operating cycle, the input to the PPC from the venturis is also adjusted using the ratio between LEFM input and the venturi input. The ratio is calculated using 30 minute averaged feed water flow data from the LEFM and the venturi at rated conditions. The 30 minute average is satisfactory to negate the effects of bi-stable core flow as discussed in Regulatory Issue Summary 2007-21, "Adherence to Licensed Power Limits." Feedwater flow input to the core thermal power calculation is provided by the existing feedwater flow venturis when LEFM data is not available. Since the feedwater flow venturis are corrected to the last validated data from the LEFM system, it is acceptable to remain at the MUR thermal power of 3544 MWt for up to 72 hours to enact LEFM system repairs. After 72 hours, actions required by the LCS will be taken to reduce power to the appropriate level.

Since the LEFM Check Plus system has two modes of operation, LCS 1.3.9 allows for an intermediate power reduction. With one or both LEFM feedwater flow meters in the Check mode (one plane out of service on one or both meters), feedwater measurement uncertainty increases from ≤ 0.3% to ≤ 0.5%. This additional uncertainty equates to a 0.2% power reduction from MUR uprate thermal power to 3537 MWt. As noted in the LCS provided, if the LEFM system is not returned to functionality within 72 hours, power will be reduced and administratively controlled to remain less than or equal to 3537 MWt. A similar allowance was approved in Reference 6.13.

The 72-hour Completion Time for the LEFM system prior to reducing to the CLTP is acceptable. As discussed above, during the 72 hour Completion Time, the existing feedwater flow venturi-based signals will be corrected to the last validated data from the LEFM system. Although the feedwater flow venturi measurement signals may drift slightly during this period due to fouling of the feedwater flow venturis, such fouling results in a higher than actual indication of feedwater flow. This condition results in an overestimation of the calculated thermal heat balance power level, which is conservative, as the reactor will actually be operating below the calculated power level. Note that the NRC has previously approved power uprate applications with Completion Times of up to 72 hours for similar BWRs (References 6.3 through 6.5 and 6.13).

Regarding potential drift in the measurement of feedwater differential pressure across the feedwater flow venturis, industry experience for similar BWRs shows that the instrument drift associated with feedwater flow measurements are insignificant over a 72 hour time period. In Reference 6.7, Table A-1 provides the systematic error associated with feed flow nozzle differential pressure as approximately 1.0% over an operating cycle. Thus, over a 72-hour period, this would have an insignificant effect on the feedwater flow measurement.

A sudden de-fouling event during the 72-hour Completion Time is unlikely. Significant sudden de-fouling would be detected by a change in the balance of plant parameters. A review of recent plant operating experience has not identified any instances of sudden de-fouling events at Columbia.

Criterion 2

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 Caldon Topical Report ER-80P.

Response to Criterion 2

The LEFMs were installed during the spring 2015 RFO. Following commissioning, the LEFM system was used to supply the feedwater flow input to the PPC core thermal power calculation and the station has remained ≤ 3486 MWt (CLTP). Since the commissioning of the LEFM, the following maintenance issues have occurred:

- An error was introduced into the LEFM transmitter configuration files due to an incorrect configuration file change provided by the vendor, Cameron. Condition reports (CRs) were initiated to correct the error. A cause evaluation was performed and determined that there were weaknesses in the configuration control and a lack of rigor in validating vendor-supplied changes to the configuration file. Actions are being taken to address the causes including instituting more robust controls on software quality and configuration. Cameron has taken actions to fully review all current configuration files and provide a comparison file with explanations for any file updates. This condition has been entered into the station's corrective action program.

- Four out of 16 Paths have experienced degraded transducer signal quality. Troubleshooting investigation points to loose wiring at the transducers as a likely cause. A work order was initiated to troubleshoot and correct the problem with the transducers when access to transducers is available. This condition has been entered into the station's corrective action program.
- The LEFM CPU has experienced lock-ups which results in stale flow data being output to the plant computer. A CR was initiated to address this issue. Preliminary reviews by the vendor, Cameron, have identified an error in the LEFM watchdog timer configuration settings. Corrective actions are in place to resolve the configuration error. Cameron also recommends periodically rebooting the CPUs to eliminate lock-ups commonly experienced on personal computers that are continuously running. Actions to create recurring tasks to reboot the CPUs quarterly are being taken. This condition has been entered into the station's corrective action program.

These issues have been discussed with Cameron, who is working with the LEFM system engineers to assess the issues and provide resolutions. Cameron has also agreed to provide a root cause report outlining all errors experienced and root causes of these errors.

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

The method used in performing this calculation is based on the accepted plant setpoint methodology Standard EES-4, Setpoint Methodology. This standard is based on the American Society of Mechanical Engineers (ASME) PTC 19.1-1985, "Measurement Uncertainty," and the Instrumentation, Systems, and Automation Society (ISA) RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The methodologies used in the heat balance determination (Enclosure 13) are discussed in Section 3.2.3 of this enclosure.

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.

<u>Response to Criterion 4</u>

This is not applicable to Columbia. The calibration factors for the Columbia ultrasonic LEFM flow meters were established by tests of these flow meters at Alden Research Laboratory. These tests were performed on a full-scale model of the Columbia hydraulic geometry. A discussion of the impact of the plant-specific installation factors on the feedwater measurement uncertainty is provided in Cameron Report ER-1049, Revision 3, (Enclosure 10) and Cameron Report ER-1074, Revision 0 (Enclosure 11). The test configurations modeled the portion of piping upstream of the LEFM spool pieces and can be compared to the plant installation drawings by comparing the drawings Enclosure 11, Figures 2.1 and 2.2, to the installation drawings in Enclosure 14. There is no significant difference between the Columbia feedwater piping configuration and the test configuration used at Alden Research Laboratory.

<u>Criterion 5</u>

Continued operation at the pre-failure power 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**.**

<u>Response to Criterion 5</u>

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

<u>Criterion 6</u>

A CheckPlus operating with a single failure is not identical to 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 an increased uncertainty.

<u>Response to Criterion 6</u>

As identified in Enclosure 10, using the total thermal power uncertainty approach documented in Reference 6.8, the uncertainty in the Columbia LEFM CheckPlus system measurement is as follows:

- Total thermal power uncertainty in the LEFM Check Plus mode is ± 0.276%.
- Total thermal power uncertainty in the LEFM Check mode is ± 0.485%.

The LEFM CheckPlus system is in Check mode when one or both LEFM system meters are in the Check mode and not in Fail. The total uncertainty of the LEFM CheckPlus system operating in the Check mode was evaluated in Enclosure 11 and resulted in the increased uncertainty stated above.

Criterion 7

An applicant with a comparable geometry can reference the Section 3.2.1 finding (of Reference 6.10) to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test 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 installation configuration of the Columbia LEFM system spool pieces are described in Section 3.2.2 of this enclosure. Testing was conducted at Alden Research Laboratories as described in Enclosure 11. The hydraulic model configuration was designed as a hydraulic duplicate of the principle hydraulic features of the installation site (ALD-1160, Hydraulic Calibration Plan for Columbia Nuclear Generating Station, Revision 2, which is Reference 1 of Enclosure 11, contains the plant details). The tests conducted at the Alden Research Laboratories verified the use of an LEFM CheckPlus system with disabled components make the CheckPlus system comparable to a Check system. The testing supports that the downstream geometry does not have a significant influence on the Columbia LEFM system calibration.

Criterion 8

An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 (of Reference 6.10) should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 (of Reference 6.10). 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 LEFM system spool pieces at Columbia are both located in the feed water lines downstream of a 90 degree elbow. The venturi flow elements are located upstream of the 90 degree elbow. A flow straightener is located at the inlet to each of the venturi flow elements. The arrangement of the 90 degree elbow, the venturi flow element and the flow straightener were all modeled in detail during testing at Alden Research Laboratories. A full range of flow tests were performed in the normal piping configuration on both LEFM meters. Flow testing was also performed by rotating the flow straightener, which indicated that it had no significant effect in the LEFM calibration. Additional flow testing was performed with inline flow disruptions, half moon plates at various locations before the flow straightener, before the venturi and before the 90 degree elbow. These flow disruptions created significantly larger flow profile asymmetry and flow swirl than existed in the normal plant piping configuration. The testing results indicated that actual increases in the flow profile asymmetry and flow swirl cause the LEFM meter to indicate a more conservative flow. Based on the results of this testing the flow straightener located upstream of the venturi is sufficiently far enough upstream of the LEFM meter that its effect does not significantly impact the operation of the LEFM in the Check Plus or Check mode.

Criterion 9

An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 of Reference 6.10.

Response to Criterion 9

Columbia conservatively assumes no moisture content in the core thermal power uncertainty calculation (Enclosure 13). This approach is consistent with that described in Section 3.2.3 of Reference 6.10. Thus, this criterion is not applicable to Columbia.

3.2.5 Deficiencies and Corrective Actions

Cameron has procedures to notify users of important LEFM deficiencies. Columbia also has processes for addressing manufacturer's deficiency reports. Such deficiencies are documented in Columbia's corrective action program. Deficiencies associated with the vendor’s processes or equipment are reported to the vendor to support corrective action.

3.2.6 Reactor Power Monitoring

Energy Northwest's Policy Statement Manual provides guidance to ensure that reactor power remains within the requirements of the operating license. Plant procedures provide requirements for monitoring and controlling reactor power in compliance with TS that is consistent with the guidance proposed by the Nuclear Energy Institute (NEI) and endorsed by the NRC in Reference 6.11.

3.3 Evaluation of Changes to Operating License and Technical Specifications

The proposed changes to the TS described in Section 2.0, “Detailed Description,” are evaluated below. The numbering of these changes corresponds to the numbering in Section 2.0.

Sections 2.1 and 2.2, Changes Related to RTP

The proposed increase in RTP in the Columbia OL and TS Definitions is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM feedwater flow measurement system and on the evaluations provided in this License Amendment Request.

Section 2.3, Changes Related to Revised Allowable Values for the Average Power Range Monitors Simulated Thermal Power – High Trip Function

The proposed changes in the two-loop and single-loop Average Power Range Monitor Simulated Thermal Power - High trip functions are contained in TS 3.3.1.1 Table 3.3.1.1-1, Function 2.b. The proposed change to the Allowable Values (AVs) for the Average Power Range Monitors Simulated Thermal Power - High Trip functions are based on the approach described in Reference 6.6, Section F.4.2.1, “Flow Referenced

APRM Trip and Alarm Setpoints." The Average Power Range Monitor Simulated Thermal Power – High trip function AVs, for both two-loop operation and single-loop operation, 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 MUR power uprate. The specific values are provided in Section 5.3, "Technical Specification Instrument Setpoints," of Enclosure 7. The AVs were generated using approved GEH setpoint methodology. Further discussion of the setpoint methodology is found in Section 3.4.4 of this enclosure.

Section 2.3, Changes Related to Revised Allowable Values for Turbine Throttle Valve - Closure and Turbine Governor Valve - Fast Closure, Trip Oil Pressure - Low

The proposed change for the power level at which the Turbine Throttle Valve - Closure and Turbine Governor Valve - Fast Closure, Trip Oil Pressure - Low trip functions are bypassed are contained in TS 3.3.1.1, Required Action E.1, SR 3.3.1.1.12, and Table 3.3.1.1-1, Functions 8 and 9. The bypass of these trip functions is accomplished by sensing turbine first-stage pressure. Based on the guidelines in Section F.4.2.3, "Turbine First-Stage Pressure Signal Setpoint," of Reference 6.6, the value at which the Turbine Throttle Valve - Closure trip and Turbine Governor Valve - Fast Closure, Trip Oil Pressure - Low trip functions are bypassed, in percent of RTP, is reduced by the ratio of the MUR power uprate increase. The value does not change with respect to absolute thermal power. The specific values are provided in Section 5.3 of Enclosure 7.

Section 2.4, Changes related to End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The proposed change to the power level at which the Turbine Throttle Valve (TTV) - Closure and Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure - Low trip functions are bypassed are contained in TS 3.3.4.1, APPLICABILITY, Required Action C.2, and SR 3.3.4.1.3. The EOC-RPT function is automatically disabled by sensing turbine first stage pressure. Based on the guidelines in Reference 6.6, Section F.4.2.3, "Turbine First-Stage Pressure Signal Setpoint," the value at which the TTV - Closure and TGV Fast Closure, Trip Oil Pressure - Low trip functions are bypassed, in percent of RTP, is reduced by the ratio of the MUR power uprate increase. The value does not change with respect to absolute thermal power. The specific values are provided in Section 5.3 of Enclosure 7.

Section 2.5, Changes related to the Primary Containment Isolation Instrumentation

The proposed change to the Main Steam Line Flow - High pressure setpoint is contained in TS 3.3.6.1, Table 3.3.6.1-1, Function 1.c. As stated in Section 5.3.5, "Main Steam Line High Flow Isolation," of Enclosure 7, a new setpoint as a result of the increased steam flow was calculated using the GEH setpoint methodology. A TS AV change is required to change the differential pressure setpoint at the allowable steam flow.

3.4 Additional Considerations

3.4.1 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 Enclosure 7 providing further detail.

Topic	**Conclusion**	**Enclosure 7 Section**
Normal Plant Operating Conditions	MUR power uprate is accommodated by increasing core flow along previously established MELLLA rod lines.	Section 1
Reactor Core and Fuel Performance	Reactor core and fuel design is adequate for operation at MUR uprated conditions.	Section 2
Reactor Coolant and Connected Systems	Overpressure protection, fracture toughness, structural, and piping evaluations are acceptable.	Section 3
Engineered Safety Features	Acceptable based on previous analyses at 102% of current licensed power.	Section 4
Instrumentation and Control	Current instrumentation is acceptable. Changes to some TS values are necessary.	Section 5
Electrical Power and Auxiliary Systems	Minor increases in normal power system loads. Emergency power systems are unaffected. Auxiliary systems are acceptable.	Section 6
Power Conversion Systems	Power conversion systems are adequate without modification.	Section 7
Radwaste and Radiation Sources	Small increases in normal operation radiation levels and effluents. Accident consequences are bounded by previous evaluations.	Section 8
Reactor Safety Performance Evaluations	Design basis events are bounded by previous evaluations. Special events meet acceptance criteria.	Section 9
Other Evaluations	All evaluation results are acceptable.	Section 10

3.4.2 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. This experience has been associated with extended power uprates (EPUs), and not with smaller uprates, such as MUR power uprates.

The generic evaluation provided in Reference 6.6, Appendix J.2.3.7 is applicable to Columbia. The requirements for the main steam isolation valves (MSIVs) remain unchanged for MUR uprate conditions. All safety and operational aspects of the MSIVs are within previous evaluations.

The stresses of the RPV internals that were affected by GEH Safety Communications were reconciled for the increase of the acoustic load to show that adequate stress margins still exist and the stresses remain within the allowable limits. All the RPV internals were shown to be within the allowable limits. The limiting stresses of all RPV internal components are summarized in Enclosure 7, Table 3-8. Therefore the RPV internal components are demonstrated to be structurally qualified for operation at MUR uprate conditions.

Based on the above, no adverse flow induced vibration effects are expected as a result of the MUR power uprate.

3.4.3 Plant Modifications

The evaluations performed to support the MUR power uprate identified that no physical modifications are required to plant systems. However, software changes to PPC are required to support the interface with the LEFM system for operation above the CLTP limit of 3486 MWt.

3.4.4 Instrument Setpoint Methodology

The determination of Allowable Values described in Section 2.0 of this enclosure is based on the GEH setpoint methodology. Reference 6.6 used approved GEH setpoint methodology to generate the values. 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.

Columbia previously adopted portions of Technical Specification Task Force (TSTF) Traveler TSTF-493, “Clarify Application of Setpoint Methodology for LSSS Functions”, Revision 4, for the Average Power Range Monitor instrumentation in Amendment 226, which was approved on January 31, 2014. This amendment added Notes (d) and (e) to TS Table 3.3.1.1-1.

3.4.5 Grid Stability Studies

The Columbia Final Safety Analysis Report (FSAR) Section 6.3.2.2 on equipment and component descriptions states the following. Regular AC power is from the main transformers [TR-N(1) and (2)] during plant operation or from the startup transformer (TR-S) (an offsite power source) when the main generator is off-line. Should regular AC power be lost, Division 1 (low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI) loop A) and Division 2 (LPCI loops B and C) would be transferred to a second offsite power supply and backup transformer (TR-B). Division 3 high pressure core spray (HPCS) would be powered from its onsite standby diesel. If the backup transformer were also lost, Divisions 1 and 2 would then be powered from their respective and independent onsite standby diesels. A more detailed description of the power supplies for the ECCS is contained in FSAR Section 8.3.

Enclosure 15 provides the grid study performed to assess the effects of the MUR power uprate. A steady-state power-flow study and a transient study were performed for specific contingencies including transfer of station service load to TR-S or TR-B offsite sources following a reactor scram. The power flow studies are comprised of a post-contingency voltage assessment and a voltage stability study. The study found no adverse effects from the additional generating capacity resulting from the uprate and that the existing system has the ability to maintain the required 1.0 p.u.(per unit) voltage at the off-site station service sources TR-S and TR-B. The transient stability studies ensure that 500-kV line faults or loss of major generation does not result in undamped conditions, voltage dip violation or frequency excursion violations in accordance with the reliability criteria. Results for the transient study cases show that each contingency was transiently stable and dynamically damped.

3.4.6 Operator Training, Human Factors, and Procedures

The operator response to plant transients or accidents is unaffected by the proposed power uprate changes. When the LEFM system status shifts from the Check Plus mode to the Check mode, the control room operators are alerted with a visual alarm from the PPC. The proposed LCS Requirement for Operation provided in Enclosure 3 provides the Required Compensatory Measures and Completion Times for the identified Conditions. These are the only new operator actions associated with this license amendment request. The PPC, with displays at the Reactor Operator and Control Room Supervisor stations, will provide a visual alarm to alert the operators to changes in the LEFM system status (See Enclosure 6, Item 6). The LEFM electronics unit installed in the system control cabinet contains a display and keyboard that is used to respond to system status changes when indicated by the PPC visual alarm.

The Plant Process Computer provides LEFM status information through the PPC Overview display. The initial indication of a change in LEFM status is immediate but non-intrusive to the operators. This ensures that the operators are aware that the status of the LEFM has changed but does not require any immediate action from the operators. The PPC includes a nominal time allowance for normal and expected operational conditions such as momentary rejection of transducer data or momentary failure of the LEFM data validation check that are resolved without operator intervention. These conditions are normally self-correcting. If the condition exists for longer than the specified time allowance, then an actual LEFM failure may exist. At this point the PPC will generate the visual alarm to notify the operator of a change in the LEFM system status. This ensures that the Human Machine Interface (HMI) of the PPC does not become an operator distraction. An audible alarm is not required since the operators routinely monitor the PPC Overview display as part of the normal duties to ensure thermal power is maintained within limits.

Necessary operating procedure revisions will be completed prior to implementation of the proposed MUR power uprate (See Enclosure 6, Item 2). The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes (See Enclosure 6, Item 3). Any necessary operator training will be completed prior to implementation of the proposed changes (See Enclosure 6, Item 4).

3.4.7 Plant Testing

Plant testing for the MUR power uprate will be completed as described in Section 10.4, “Testing” of Enclosure 7 (See Enclosure 6, Item 5).

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 31, 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 Precedents

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

Facility	Amendment No(s).	Approval Date	Accession No.
LaSalle, Units 1 and 2*	198/185	September 16, 2010	ML101830361
Limerick, Units 1 and 2*	201/163	April 8, 2011	ML110691095
Fermi 2* Correction	196	February 10, 2014 March 14, 2014	ML13364A131 ML14066A410
Shearon Harris*	139	May 30, 2012	ML11356A096

* CheckPlus system

Unlike this Columbia submittal, the listed precedent submittals of LaSalle, Limerick and Fermi also included a request that included TSTF-493, “Clarify Application of Setpoint Methodology for LSSS Functions”, Revision 4. Columbia incorporated portions of TSTF-493 as discussed in Section 3.4.4, Instrument Setpoint Methodology.

Similar to the approved Shearon Harris submittal, Columbia is also proposing use of the Check mode allowing the use of an increased uncertainty allowing operation at power level greater than the CLTP, but less than MUR uprated power as discussed in Section 3.2.1 of this enclosure.

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 Models,” Energy Northwest requests an amendment to Columbia Generating Station (Columbia) Renewed Facility Operating License (OL) NPF-21. Specifically, the proposed changes revise the OL and Technical Specifications (TS) to implement an increase of approximately 1.66% in RTP from 3486 megawatts thermal (MWt) to 3544 MWt. These changes are based on increased feedwater measurement accuracy, which was achieved by utilizing Cameron International (formerly Caldon) CheckPlus Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

According to 10 CFR 50.92, “Issuance of Amendment,” paragraph (c), a proposed amendment to an operating license does not involve a significant hazard if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Energy Northwest has evaluated the proposed changes, using the criteria in 10 CFR 50.92, and has determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

Response: No

The proposed change will increase the Columbia Generating Station rated thermal power from 3486 MWt to 3544 MWt. The reviews and evaluations performed to support the proposed uprated power conditions included all structures, systems and components that would be affected by the proposed changes. The reviews and evaluations determined that these structures, systems, and components are capable of performing their design function at the proposed uprated RTP of 3544 MWt. All accident mitigation systems will function as designed, and all performance requirements for these systems have been evaluated and were found acceptable. Thus, the proposed changes do not create any new accident initiators or increase the probability of an accident previously evaluated.

The primary loop 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. Thus, there is no increase in the probability of a structural failure of these components.

The nuclear steam supply systems will continue to perform their intended design functions during normal and accident conditions. 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, Energy Northwest has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions.

The current safety analyses remain applicable, since they were performed at power levels that bound operation at a core power of 3544 MWt. The results demonstrate that acceptance criteria of the applicable analyses continue to be met at the uprated conditions. As such, all applicable accident analyses continue to comply with the relevant event acceptance criteria. 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.

Power level is an input assumption to equipment design and accident analyses, but it is not a transient or accident initiator. Accident initiators are not affected by power uprate, and plant safety barrier challenges are not created by the proposed changes. 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

No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. All structures, systems 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.

Plant operation at a RTP of 3544 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analysis of record or recent evaluations demonstrate that applicable criteria are still met with the proposed changes. 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 margins of safety associated with the power uprate are those pertaining to core thermal power. 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

Based on the above evaluation, Energy Northwest 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.

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.

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 uprated power 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.

There is no significant change in the types or significant increase in the amounts of any effluents. Evaluations of the effects of the proposed changes on effluent sources concluded that the increase in effluents will be small, and within the current applicable permits and regulations.

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 power 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 is required in connection with the proposed amendment.

6.0 References

6.1 Columbia NRC Docket No. 50-397 NRC License No. NPF-21

6.2 NRC Regulatory Issue Summary 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” dated January 31, 2002 (ML013530183)

6.3 Letter from Carl F. Lyon (USNRC) to Stewart B. Minahan (Nebraska Public Power District), “Cooper Nuclear Station – Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC No. MD7385),” dated June 30, 2008 (ML081540280)

6.4 Letter from Christopher Gratton (USNRC) to Michael J. Pacilio (Exelon Nuclear), “LaSalle County Station, Units 1 and 2 – Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME3288 and ME3289),” dated September 16, 2010 (ML101830361)

6.5 Letter from Peter Bamford (USNRC) to Michael J. Pacilio (Exelon Nuclear). “Limerick Generating Station, Units 1 and 2 – Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate and Standby Liquid Control System Changes (TAC Nos. ME3589, ME3590, ME3591, and ME3592),” dated April 8, 2011 (ML110691095)

6.6 General Electric-Hitachi (GEH) Nuclear Energy Report NEDC-32938P-A, “Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” Revision 2, dated May 2003

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

6.8 Caldon Topical Report ER-157P, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System,” Revision 8, dated May 2008

6.9 Letter from John N. Hannon (USNRC) to C. Lance Terry (TU Electric), “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,’ (TACS Nos. MA2298 and MA2299),” dated March 8, 1999 (ML9903190053)

6.10 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 a Power Uprate With the LEFM Check or CheckPlus System,’ (TAC No. ME1321),” dated August 16, 2010 (ML102160663)

6.11 Memorandum from Timothy Kolb (USNRC) to Mike Case (USNRC), “Safety Evaluation Regarding Endorsement of NEI Guidance for Adhering to the Licensed Thermal Power Limit (TAC No. MD9233),” dated October 8, 2008 (ML082690105)

6.12 Cameron Manual IB1404 “LEFM CheckPlus Flow Measurement System Installation and Commissioning Manual for Columbia Nuclear Power Plant”, Revision 0, dated March 2014

6.13 Letter form Araceli T. Billoch Colon (USNRC) to Chris Burton (Progress Energy Carolinas, Inc.) “Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC NO. ME6169)”, dated May 30, 2012 (ML11356A096)

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 2

Markup of Existing Renewed Facility Operating License and Technical Specifications

Renewed Facility Operating License

Page 3

Technical Specifications Pages

1.1-5

3.3.1.1-10

3.3.1.1-13

3.3.1.1-15

3.3.1.1-18

3.3.4.1-1

3.3.4.1-2

3.3.4.1-3

3.3.6.1-5

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

(6) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to store byproduct, source and special nuclear materials not intended for use at Columbia Generating Station. The materials shall be no more than 9 sealed neutron radiation sources designed for insertion into pressurized water reactors and no more than 40 sealed beta radiation sources designed for use in area radiation monitors. The total inventory shall not exceed 24 microcuries of strontium-90, 20 microcuries of uranium-235, 30 curies of plutonium-238, and 3 curies of americium-241.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (~~3486~~3544 megawatts thermal).

1.1 Definitions

PHYSICS TESTS (continued)	
	c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3486~~ 3544 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
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.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more 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 < ~~30~~29.5% 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
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Deleted.	
SR 3.3.1.1.10	------------------------------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.	 18 months for Functions 1, 3, 4, 6, 7, and 9 through 11 <u>AND</u> 24 months for Functions 2, 5, and 8
SR 3.3.1.1.11	Deleted.	
SR 3.3.1.1.12	Verify Turbine Throttle Valve - Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 29.5~~30~~% RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months

Table 3.3.1.1-1 (page 1 of 4)
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. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[(a)]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High (Setdown)	2	3[(b)]	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10[(d),(e)] SR 3.3.1.1.16	≤ 20% RTP
b. Simulated Thermal Power - High	1	3[(b)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10[(d),(e)] SR 3.3.1.1.16	≤ 0.6~~3~~2W + 6~~4~~2.~~0~~9% RTP and ≤ 114.9% RTP[(c)]

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

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(c) ≤ 0.6~~3~~2W + ~~60~~59.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(d) 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.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The LTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Licensee Controlled Specifications.

Table 3.3.1.1-1 (page 4 of 4)
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
8. Turbine Throttle Valve - Closure	≥ 29.5~~30~~% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed
9. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≥ ~~30~~29.5% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1000 psig
10. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
	5[(a)]	2	H	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

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

3.3 INSTRUMENTATION

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

LCO 3.3.4.1

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

1. Turbine Throttle Valve (TTV) – Closure; and

2. Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~30~~29.5% 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. OR A.2 ---------------NOTE-------------- Not applicable if inoperable channel is the result of an inoperable breaker. ------------------------------------- Place channel in trip.	72 hours 72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	OR		
	B.2	Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Remove the associated recirculation pump from service.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to < ~~30~~29.5% 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 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.2.a	Perform CHANNEL CALIBRATION. The Allowable Value shall be: TTV - Closure: ≤ 7% closed.	24 months
SR 3.3.4.1.2.b	Perform CHANNEL CALIBRATION. The Allowable Value shall be: TGV Fast Closure, Trip Oil Pressure - Low: ≥ 1000 psig.	18 months
SR 3.3.4.1.3	Verify TTV – Closure and TGV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ ~~30~~29.5% RTP.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.5	------------------------------NOTE----------------------------- Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. --- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.4.1.6	Determine RPT breaker arc suppression time.	60 months

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -142.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow - High	1, 2, 3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 137~~24~~.9~~4~~ psid
d. Condenser Vacuum - Low	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1, 2, 3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1, 2, 3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1, 2, 3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches

(a) With any turbine throttle valve not closed.

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 3

New Licensee Controlled Specification and Markup of Technical Specification Bases "For Information Only"

Licensee Controlled Specifications Pages

1.3.9-1

1.3.9-2

B 1.3.9-1 through B 1.3.9-5

Technical Specifications Bases Pages

B 3.3.1.1-21

B 3.3.1.1-22

B 3.3.1.1-35

B 3.3.2.2-1

B 3.3.4.1-2

B 3.3.4.1-3

B 3.3.4.1-4

B 3.3.4.1-6

B 3.3.4.1-7

B 3.7.6-1

1.3 INSTRUMENTATION

1.3.9 Leading Edge Flow Meter (LEFM) Feedwater Flow Instrumentation

RFO 1.3.9 The LEFM Feedwater Flow Instrumentation System shall be OPERABLE.

APPLICABILITY: THERMAL POWER > 3486 MWt.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Loss of LEFM Meter Status indication.	A.1 Restore LEFM Meter Status indication.	72 hours
B. Required Compensatory Measure and associated Completion Time of Condition A not met.	B.1 Reduce power to ≤ 3486 MWt.	Immediately
C. One or more LEFM feedwater flow meters not in the Check Plus Mode.	C.1 --------------NOTE-------------- If current THERMAL POWER is < 3537 MWt, the maximum permissible THERMAL POWER is 3537 MWt. ------------------------------------- Return both LEFM feedwater flow meters to the Check Plus Mode.	72 hours

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
D. Required Compensatory Measure and associated Completion Time of Condition C not met.	D.1 Reduce THERMAL POWER to ≤ 3537 MWt. <u>AND</u> D.2 --------------NOTE-------------- Not applicable if one or more LEFM feedwater flow meters are in the Fail Mode. ------------------------------------- Verify both LEFM feedwater flowmeters are in the Check Mode or one LEFM feedwater flowmeter is in the Check Mode and one LEFM feedwater flowmeter is in the Check Plus Mode.	Immediately Once per 24 hours
E. One or more LEFM feedwater flow meters in the Fail Mode.	E.1 Reduce THERMAL POWER to ≤ 3486 MWt.	72 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 1.3.9.1	----------------------------NOTE------------------------------ LEFM feedwater flow meter status is monitored by the plant process computer, which will alarm when the LEFM System is determined to be not in the Check Plus Mode. --- Verify each LEFM feedwater flow meter is in the Check Plus Mode.	24 hours

B 1.3 INSTRUMENTATION

B 1.3.9 Leading Edge Flow Meter (LEFM) Feedwater Flow Instrumentation

BASES

BACKGROUND

The Leading Edge Flow Meters (LEFM) provides improved accuracy compared to the feedwater flow venturis when used in the heat balance calculation of THERMAL POWER. The LEFM allows an increase in THERMAL POWER from 3486 MWt to 3544 MWt. The feedwater flow venturis are periodically corrected for drift to match the indication of the LEFM and are used as backup to the LEFM in the event that the LEFM fails. One LEFM is installed on each feedwater flow line.

The LEFM System continuously performs online self-diagnostics to verify the system operation is within design basis uncertainty limits. Any out-of-specification condition for either LEFM will result in a self-diagnostic alarm condition. Each LEFM consists of two measurement planes 90° apart. Each plane consists of four acoustic paths made up of two transducers per path. The LEFM is operating in the Check Plus Mode when all acoustic paths are in operation. A failed transducer in one measurement plane will cause the affected measurement plane to be nonfunctional. The LEFM is operating in the Check Mode when one of the two measurement planes is nonfunctional. The LEFM is considered failed if both measurement planes are nonfunctional.

If the communications link between the LEFM System and the Plant Process Computer (PPC) fails (i.e., LEFM CPU Link A and B failed), the LEFM is considered nonfunctional.

LEFM Mode of Operation	Max THERMAL POWER
Check Plus Mode	3544 MWt
Check Mode	3537 MWt
Fail Mode	3486 MWt
Communication Failure / Loss of Indication	3486 MWt

APPLICABLE SAFETY ANALYSES

No specific safety analyses take direct credit for the LEFM feedwater flow instrumentation. However, RATED THERMAL POWER plus uncertainty is an initial condition for many design basis accidents and transients.

This function indirectly ensures that THERMAL POWER does not exceed the assumed initial conditions in the safety analyses.

REQUIREMENTS FOR OPERABILITY

Both LEFM feedwater flow meters are required to be OPERABLE. LEFM feedwater flow meter status is monitored by the PPC, which will alarm when the LEFM System is determined to be not in the Check Plus Mode.

BASES

APPLICABILITY	THERMAL POWER > 3486 MWt.

COMPENSATORY MEASURES

A.1

When the PPC data link fails (total loss of communication between both PPC CPUs and both LEFM CPUs) the LEFM meter status indication must be restored. On loss of data link, the PPC initiates automatic actions to restore the connection and core thermal power calculations revert to using calibrated venturi inputs.

A Completion Time of 72 hours from the point that a valid loss of signal is confirmed is reasonable because the feedwater flow venturis are periodically calibrated using the LEFM instrumentation and subsequent venturi drift is small over the Completion Time.

If the cause of the loss of data link between the PPC and LEFM is determined to be due to actual LEFM failure, then Condition E should be entered immediately.

B.1

With Required Compensatory Measure A.1 not met, Required Compensatory Measure B.1 requires that THERMAL POWER be immediately reduced to less than or equal to 3486 MWt. In this Condition, THERMAL POWER uncertainty increases to 2% of 3486 MWt based upon the accuracy of the feedwater flow venturis (Reference 1). Therefore, THERMAL POWER is reduced to 3486 MWt to ensure that the initial conditions of the safety analyses remain valid. At this point, 3486 MWt is the new maximum THERMAL POWER limit.

C.1, D.1, D.2 and E.1

With one or more LEFM feedwater flow meters not in the Check Plus Mode, Required Compensatory Measure C.1 requires that the affected LEFM feedwater flow meter(s) be restored to the Check Plus Mode.

BASES

COMPENSATORY MEASURES (continued)

If both LEFM feedwater flow meters are in the Check Mode or one LEFM feedwater flow meter is in the Check Mode and one LEFM feedwater flow meter is in the Check Plus Mode, the allowed Completion Time of 72 hours is reasonable since the LEFM feedwater flow instrumentation remains functional in this Condition and allows time for maintenance on the LEFM instrumentation system.

If one or more LEFM feedwater flow meters are in the Fail Mode, the allowed Completion Time of 72 hours is reasonable because the feedwater flow venturis are periodically calibrated using the LEFM instrumentation and subsequent venturi drift is small over the Completion Time. Note that if one or more LEFM feedwater flow meters are in the Fail Mode, Condition E is also entered concurrently. Required Compensatory Measure E.1 requires reduction in THERMAL POWER within 72 hours. Thus, either the LEFM feedwater flow meters are restored per Required Compensatory Measure C.1 or THERMAL POWER is reduced per Required Compensatory Measure E.1 within 72 hours of the LEFM feedwater flow meters in the Fail Mode.

Required Compensatory Measure C.1 is modified by a Note that limits the maximum permissible THERMAL POWER to less than or equal to 3537 MWt. This note addresses the situation when one or more LEFM feedwater flow meters are not in the Check Plus Mode at reduced power levels. This note prohibits returning to RATED THERMAL POWER while in this Condition.

Conditions C and D are structured to ensure that actions are taken within 72 hours of the initial occurrence of an LEFM feedwater flowmeter not in the Check Plus Mode.

Thus, with Required Compensatory Measure C.1 not met, Required Compensatory Measure D.1 requires that THERMAL POWER be immediately reduced. If only Conditions C and D are met and Condition E is not met, then the LEFM feedwater flow meters are either both in the Check Mode or one is in the Check Mode and one is in the Check Plus Mode. Neither LEFM feedwater flow meter is in the Fail Mode. In this case, power must be reduced to less than or equal to 3537 MWt.

With both LEFM feedwater flow meters in the Check Mode or with one LEFM feedwater flow meter in the Check Mode and one LEFM feedwater flow meter in the Check Plus Mode, LEFM uncertainty increases from 0.3% to 0.5% (Reference 2). Therefore, THERMAL POWER must be reduced to 3537 MWt (Reference 3) to ensure that the initial conditions of the safety analyses remain valid. At this point, 3537 MWt is the new maximum THERMAL POWER limit.

BASES

COMPENSATORY MEASURES (continued)

Required Compensatory Measure D.2 requires that the LEFM feedwater flow meters be verified to be in either the Check Mode or the Check Plus Mode on a periodic basis. This frequency is reasonable because the LEFM System performs online self-diagnostics to verify that the system operation is within design basis uncertainty limits. Any out-of-specification condition will result in a self-diagnostic alarm condition, either for "alert" status (i.e., increased flow measurement uncertainty) or "failure" status. Required Compensatory Measure D.2 is modified by a Note stating the action is not applicable with one or more LEFM feedwater flow meters in the Fail Mode.

Required Compensatory Measure E.1 requires that with one or more LEFM feedwater flow meters in the Fail Mode, THERMAL POWER must be reduced to less than or equal to 3486 MWt. When one or more LEFM feedwater flow meters are in the Fail Mode, LEFM flow uncertainty cannot be guaranteed (Reference 2). Therefore, THERMAL POWER must be reduced to 3486 MWt (Reference 3) to ensure that the initial conditions of the safety analyses remain valid. At this point, 3486 MWt is the new maximum THERMAL POWER limit.

The allowed Completion Time of 72 hours is measured from the time that one or more LEFM feedwater flow meters enter the Fail Mode. Note that the allowed Completion Time of Condition C may already be expired if entering the Fail Mode from the Check Mode.

Within 72 hours of this Condition being met, THERMAL POWER must be reduced to 3486 MWt. The allowed Completion Time is reasonable to allow time for maintenance on the LEFM instrumentation system and because the feedwater flow venturis are periodically calibrated using the LEFM instrumentation and subsequent venturi drift is small over the Completion Time.

BASES

SURVEILLANCE REQUIREMENTS	SR 1.3.9.1 Each LEFM feedwater flow meter must be verified to be in the Check Plus Mode as indicated by the PPC once every 24 hours. This frequency is reasonable because the LEFM System performs online self-diagnostics to verify that the system operation is within design basis uncertainty limits. Any out-of-specification condition will result in a self-diagnostic alarm condition, either for "alert" status (i.e., increased flow measurement uncertainty) or "failure" status. Additionally, if the communications link between the LEFM System and the plant computer fails (i.e., LEFM CPU Link A and B failed), the LEFM flow meter is considered inoperable. This SR is modified by a Note which states that the LEFM feedwater flow meter status is monitored by the PPC, which will alarm when the LEFM System is determined to be not in the Check Plus Mode.
REFERENCES	1. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002. 2. Cameron (Caldon) document ER-1049, "Bounding Uncertainty Analysis for Thermal Power Determination at Columbia Nuclear Generating Station Using the LEFM √+ System, " Revision 3 (Proprietary), dated December 2015. 3. Heat balance calculation NE-02-15-08, Rev 0, Heat Balance Determination for Rated Thermal Power.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

This Function must be enabled at THERMAL POWER ≥ ~~30~~29.5% RTP. This is accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Throttle Valve - Closure Allowable Value is selected to detect imminent TTV closure thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Throttle 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 TTVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~29.5% RTP. This Function is not required when THERMAL POWER is < ~~30~~29.5% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TGVs 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 TGV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 5. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the digital-electro hydraulic fluid pressure at each governor valve. There is one pressure switch associated with each governor valve, the signal from each switch being assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~30~~29.5% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function. The basis for the setpoint of this automatic bypass is identical to that described for the Turbine Throttle Valve - Closure Function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TGV fast closure.

Four channels of Turbine Governor 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 ≥ ~~30~~29.5% RTP. This Function is not required when THERMAL POWER is < ~~30~~29.5% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels (one from each of the four independent banks of contacts), each of which inputs into one of the RPS logic channels.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

Four channels of Reactor Mode Switch - Shutdown Position Function, with two channels in each trip system, are available and required to be OPERABLE. The Reactor Mode - Switch Shutdown Position Function is required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.11 – Not Used

SR 3.3.1.1.12

This SR ensures that scrams initiated from the Turbine Throttle Valve - Closure and Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30~~29.5% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the Allowable Value and 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 an in-service calibration at THERMAL POWER ≥ ~~30~~29.5% RTP to ensure that the calibration is valid.

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~30~~29.5% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Throttle Valve - Closure and Turbine Governor 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 Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance was performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level, Level 8 reference point, causing the trip of the two feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High, Level 8 signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level - High, Level 8 instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. The channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs a main feedwater and main turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the throttle valves protects the turbine from damage due to water entering the turbine.

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 Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above ~~30~~29.5% 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 Reference 2.

LCO

The LCO requires three channels of the Reactor Vessel Water Level - High, Level 8 instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Level 8 signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. 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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TTV - Closure and the TGV 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 increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that assume EOC-RPT, are summarized in References 2 and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of initial closure movement of either the TTVs or the TGVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than does a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT as specified in the COLR are sufficient to mitigate pressurization transient effects. The EOC-RPT function is automatically disabled when THERMAL POWER, as sensed by turbine first stage pressure, is < 29.5~~30~~% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of Reference 4.

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel 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 3.3.4.1.2. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. 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. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. 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., TGV digital-electro hydraulic (DEH) pressure), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for process and all instrument uncertainties, except drift and calibration. The trip setpoints are derived

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

from the analytic limits, corrected for process and all instrument uncertainties, including drift and calibration. The trip setpoints derived in this manner provide adequate protection because all instrumentation uncertainties and process effects are taken into account.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternately, since this instrumentation protects against a MCPR SL violation with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the condition EOC-RPT inoperable is specified in the COLR.

Turbine Throttle Valve - Closure

Closure of the TTVs and a main turbine trip result 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 TTV - Closure in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring the MCPR SL is not exceeded during the worst case transient.

Closure of the TTVs is determined by measuring the position of each throttle valve. While there are two separate position switches associated with each throttle valve, only the signal from one switch for each TTV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TTV - Closure Function is such that two or more TTVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~30~~29.5% 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. Four channels of TTV - 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 TTV - Closure Allowable Value is selected to detect imminent TTV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ ~~30~~29.5% RTP. Below ~~30~~29.5% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

TGV Fast Closure, Trip Oil Pressure - Low

Fast closure of the TGVs 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 TGV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases 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 TGVs is determined by measuring the DEH 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 TGV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TGVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~30~~29.5% 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. Four channels of TGV 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 TGV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TGV fast closure.

This protection is required consistent with the analysis, whenever the THERMAL POWER is ≥ ~~30~~29.5% RTP. Below ~~30~~29.5% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Neutron Flux - High Functions of the RPS are adequate to maintain the necessary safety margins. The turbine first stage pressure/reactor power relationship for the setpoint of the automatic enable is identical to that described for TTV closure.

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

BASES

ACTIONS (continued)

on a valid signal and both recirculation pumps can be tripped. This requires two channels of the Function, in the same trip system, to each be OPERABLE or in trip, and the associated drive motor breakers to be OPERABLE or in trip. Alternatively, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

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 the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2, 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, THERMAL POWER must be reduced to < ~~30~~29.5% RTP within 4 hours. Alternately, 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 < ~~30~~29.5% RTP from full power conditions in an orderly manner and without challenging plant systems.

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.1

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

The Frequency of 92 days is based on reliability analysis (Ref. 5).

SR 3.3.4.1.2

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 plant specific setpoint methodology.

The Frequency for SR 3.3.4.1.2.b is based upon the assumption of an 18 month calibration interval, in the determination of the magnitude of equipment drift in the setpoint analysis.

A Frequency of 24 months is assumed for SR 3.3.4.1.2.a because the TTV position switches are not susceptible to instrument drift.

SR 3.3.4.1.3

This SR ensures that an EOC-RPT initiated from the TTV - Closure and TGV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30~~29.5% 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 an in-service calibration at THERMAL POWER ≥ ~~30~~29.5% RTP to ensure that the calibration is valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~30~~29.5% RTP either due to open main turbine bypass valves or other reasons), the affected TTV - Closure and TGV 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 considered OPERABLE.

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 ~~25~~23.35% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a four valve manifold connected to the main steam lines between the main steam isolation valves and the turbine throttle valves. Each of these valves is sequentially operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Digital-Electro Hydraulic Control System, as discussed in the FSAR, Section 7.7.1.5 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the valve manifold, through connecting piping, to the pressure-reducing perforated pipes located in the condenser shell.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the design basis feedwater controller failure, maximum demand event, described in the FSAR, Section 15.1.2 (Ref. 2). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of Reference 3.

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, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation.

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR.

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 4

Revised (Clean) Renewed Facility Operating License and Technical Specifications Pages

Renewed Facility Operating License

Page 3

Technical Specifications Pages

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3.3.1.1-18

3.3.4.1-1

3.3.4.1-2

3.3.4.1-3

3.3.6.1-5

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

(6) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to store byproduct, source and special nuclear materials not intended for use at Columbia Generating Station. The materials shall be no more than 9 sealed neutron radiation sources designed for insertion into pressurized water reactors and no more than 40 sealed beta radiation sources designed for use in area radiation monitors. The total inventory shall not exceed 24 microcuries of strontium-90, 20 microcuries of uranium-235, 30 curies of plutonium-238, and 3 curies of americium-241.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3544 megawatts thermal).

1.1 Definitions

PHYSICS TESTS (continued)	
	c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3544 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
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.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more 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 < 29.5% 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
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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Deleted.	
SR 3.3.1.1.10	------------------------------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.	18 months for Functions 1, 3, 4, 6, 7, and 9 through 11 <u>AND</u> 24 months for Functions 2, 5, and 8
SR 3.3.1.1.11	Deleted.	
SR 3.3.1.1.12	Verify Turbine Throttle Valve - Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is $\geq$ 29.5% RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months

Table 3.3.1.1-1 (page 1 of 4)
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. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[(a)]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High (Setdown)	2	3[(b)]	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10[(d),(e)] SR 3.3.1.1.16	≤ 20% RTP
b. Simulated Thermal Power - High	1	3[(b)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10[(d),(e)] SR 3.3.1.1.16	≤ 0.62W + 62.9% RTP and ≤ 114.9% RTP[(c)]

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

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(c) ≤ 0.62W + 59.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(d) 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.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The LTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Licensee Controlled Specifications.

Table 3.3.1.1-1 (page 4 of 4)
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
8. Turbine Throttle Valve - Closure	≥ 29.5% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed
9. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≥ 29.5% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1000 psig
10. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
	5[(a)]	2	H	SR 3.3.1.1.13 SR 3.3.1.1.14	 NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

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

3.3 INSTRUMENTATION

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

LCO 3.3.4.1

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

1. Turbine Throttle Valve (TTV) – Closure; and
2. Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 29.5% 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. OR A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. ----- Place channel in trip.	72 hours 72 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	OR		
	B.2	Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Remove the associated recirculation pump from service.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to < 29.5% 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 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.4.1.2.a	Perform CHANNEL CALIBRATION. The Allowable Value shall be: TTV - Closure: ≤ 7% closed.	24 months
SR 3.3.4.1.2.b	Perform CHANNEL CALIBRATION. The Allowable Value shall be: TGV Fast Closure, Trip Oil Pressure - Low: ≥ 1000 psig.	18 months
SR 3.3.4.1.3	Verify TTV – Closure and TGV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 29.5% RTP.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.5	------------------------------NOTE----------------------------- Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. -- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.4.1.6	Determine RPT breaker arc suppression time.	60 months

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1, 2, 3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -142.3 inches
b. Main Steam Line Pressure - Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 804 psig
c. Main Steam Line Flow - High	1, 2, 3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 137.9 psid
d. Condenser Vacuum - Low	1, 2[(a)], 3[(a)]	2	D	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 7.2 inches Hg vacuum
e. Main Steam Tunnel Temperature - High	1, 2, 3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
f. Main Steam Tunnel Differential Temperature - High	1,2,3	2	D	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 90°F
g. Manual Initiation	1, 2, 3	4	G	SR 3.3.6.1.6	NA
2. Primary Containment Isolation					
a. Reactor Vessel Water Level - Low, Level 3	1, 2, 3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ 9.5 inches

(a) With any turbine throttle valve not closed.

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 5

Regulatory Issue Summary (RIS) 2002-03 Cross Reference

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

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	Enclosure 1	3.1 3.2	Background and General Approach LEFM Flow Measurement and Core Thermal Power Uncertainty
I.1.A	NRC approval of topical report on flow measurement technique	Enclosure 1	3.2.1	LEFM Flow Measurement
I.1.B	Reference to NRC's approval of proposed measurement technique	Enclosure 1	3.2.1	LEFM Flow Measurement
I.1.C	Plant Implementation	Enclosure 1	3.2.2	Plant Implementation
I.1.D	Disposition of NRC criteria	Enclosure 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	Enclosure 1 Enclosure 13	3.2.3	LEFM and Core Thermal Power Measurement Uncertainty and Methodology Core Thermal Power Uncertainty Calculation
I.1.F	Calibration and maintenance	Enclosure 1	3.2.4 3.2.5	Disposition of NRC Criteria for Use of LEFM Topical Reports Deficiencies and Corrective Actions
I.1.G	Proposed outage time for LEFM and basis for selected time	Enclosure 1 Enclosure 3	3.2.3	LEFM and Core Thermal Power Measurement Uncertainty and Methodology Markup of Proposed Technical Requirements Manual Pages
I.1.H	Proposed actions if outage time is exceeded, and basis for actions	Enclosure 1 Enclosure 3	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports Markup of Proposed Technical Requirements Manual Pages

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

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	Enclosure 7	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	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations
III.2	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations
III.3	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations

IV. Mechanical/Structural/Material Component Integrity and Design

IV.1.A.i	Reactor vessel, nozzles, supports	Enclosure 7	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	Enclosure 7	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
		Enclosure 1	3.4.2	Adverse Flow Effects
IV.1.A.iii	Control rod drive mechanisms	Enclosure 7	2.5	Reactivity Control

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	Enclosure 7	3.4 3.5 3.5.1 3.6 3.7 3.8 3.9 3.10 3.11	Flow-Induced Vibration Piping Evaluation Reactor Coolant Pressure Boundary Piping Reactor Recirculation System Main Steam Line Flow Restrictors Main Steam Isolation Valves Reactor Core Isolation Cooling Residual Heat Removal System Reactor Water Cleanup System
IV1.A.v	Balance of plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, and containment systems)	Enclosure 7	3.5 3.5.2 6.4.1 6.4.3	Piping Evaluation Balance-of-Plant Piping Evaluation Service Water Systems Reactor / Safety Auxiliaries Closed Cooling Water System
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	N/A	N/A	N/A
IV.1.A.vii	Reactor coolant pumps	N/A	N/A	N/A
IV.1.A.viii	Pressurizer shell, nozzles, and surge lines	N/A	N/A	N/A

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.ix	Safety-related valves	Enclosure 7	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	Enclosure 7	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	Enclosure 7	3.2.2	Reactor Vessel Structural Evaluation
IV.1.B.iii	Flow induced vibration	Enclosure 7	3.4.	Flow-Induced Vibration
		Enclosure 1	3.4.2	Adverse Flow Effects

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.iv	Changes in temperature (pre-and post-uprate)	Enclosure 7	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.v	Changes in pressure (pre-and post-uprate)	Enclosure 7	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vi	Changes in flow rate (pre-and post-uprate)	Enclosure 7	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vii	High-energy line break locations	Enclosure 7	10.1 10.1.1 10.1.2	High Energy Line Break Steam Line Breaks Liquid Line Breaks

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.viii	Jet impingement and thrust forces	Enclosure 7	10.1 10.1.1 10.1.2 10.1.2.7	High Energy Line Break Steam Line Breaks Liquid Line Breaks Pipe Whip and Jet Impingement
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	Enclosure 7	3.1	Nuclear System Pressure Relief / Overpressure Protection
IV.1C.ii	Reactor vessel fluence evaluation	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure temperature limit curves	Enclosure 7	3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.iv	Reactor vessel low temperature overpressure protection	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.v	Reactor vessel upper shelf energy	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.D	Code of record	Enclosure 7	3.2 3.2.2 3.5 3.5.1 3.5.2	Reactor Vessel Reactor Vessel Structural Evaluation Piping Evaluation Reactor Coolant Pressure Boundary Piping Balance-of-Plant Piping Evaluation

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.E	Component inspection / testing programs and erosion / corrosion programs	Enclosure 7	3.5 3.5.1 3.5.2 10.6	Piping Evaluation Reactor Coolant Pressure Boundary Piping Balance-of-Plant Piping Evaluation Plant Life
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	N/A	N/A	N/A

V. Electrical Equipment Design

V.1.A	Emergency diesel generators	Enclosure 7	6.1 6.1.2	AC Power On-Site Power
V.1.B	Station blackout equipment	Enclosure 7	9.3.2	Station Blackout
V.1.C	Environmental qualification of electrical equipment	Enclosure 7	10.3 10.3.1	Environmental Qualification Electrical Equipment
V.1.D	Grid stability	Enclosure 1 Enclosure 7	3.4.5 6.1 6.1.1	Grid Studies AC Power Off-Site Power

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

VI. System Design

Section	Description	Document	Section	Title / Description
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling)	Enclosure 7	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
VI.1.B	Containment systems	Enclosure 7	4.1	Containment System Performance
			4.1.1	Generic Letter 89-10 Program
			4.1.3	Generic Letter 95-07 Program
			4.1.4	Generic Letter 96-06
V.1.C	Safety-related cooling water systems	Enclosure 7	6.4	Water Systems
			6.4.1	Service Water Systems
			6.4.4	Ultimate Heat Sink

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
V.1.D	Spent fuel pool storage and cooling systems	Enclosure 7	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
V.1.E	Radioactive waste systems	Enclosure 7	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
V.1.F	Engineered safety features (ESFs) heating, ventilation, and air conditioning systems	Enclosure 7	4.4	Main Control Room Atmosphere Control System
			6.6	Power Dependent Heating, Ventilation and Air Conditioning

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

VII. Other

Section	Description	Document	Section	Title / Description
VII.1	Operator actions and effects on time available	Enclosure 7	4.1 6.7 9.3 10.5	Containment System Performance Fire Protection Special Events Operator Training and Human Factors
VII.2.A	Emergency and abnormal operating procedures	Enclosure 7	10.9	Emergency Operating Procedures
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	Enclosure 1 Enclosure 7 Enclosure 6	3.2.4 3.4.3 10.5 #5	Disposition of NRC Criteria for Use of LEFM Topical Reports Plant Modifications Operator Training and Human Factors
VII.2.C	The control room plant reference simulator	Enclosure 7 Enclosure 6	10.5 #4	Operator Training and Human Factors
VII.2.D	The operator training program	Enclosure 7 Enclosure 6	10.5 #5	Operator Training and Human Factors
VII.3	Modifications completion	Enclosure 1	3.4.3	Plant Modifications
VII.4	Procedure Revisions – Licensed Power Level	Enclosure 1	3.2.6	Reactor Power Monitoring
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	Enclosure 1 Enclosure 7	5.0 6.4.2.1 8.6	Environmental Consideration Discharge Limits Normal Operation Off-Site Doses

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		COLUMBIA RESPONSE		
NRC RIS 2002-03		Columbia MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
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	Enclosure 1 Enclosure 7	5.0 8.5	Environmental Consideration Radiation Levels

VIII. Changes to Technical Specifications, Protection System Settings, and Emergency System Settings

Section	Description	Document	Section	Title / Description
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	Enclosure 1 Enclosure 2	1.0 2.0	Description Detailed Discussion Markup of Proposed Operating License and Technical Specifications Pages
VIII.1.A	Description of the change	Enclosure 1 Enclosure 2	1.0 2.0	Description Detailed Discussion Markup of Proposed Operating License and Technical Specifications Pages
VIII.1.B	Identification of analyses affected by and/or supporting the change	Enclosure 1 Enclosure 7	3.3	Evaluation of Changes to Operating License and Technical Specifications GEH Nuclear Energy Safety Analysis Report NEDC-33853P
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	Enclosure 1 Enclosure 7	3.3	Evaluation of Changes to Operating License and Technical Specifications GEH Nuclear Energy Safety Analysis Report NEDC-33853P

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 6

Summary of Regulatory Commitments

Enclosure 6

Summary of Regulatory Commitments

	COMMITMENT	COMMITED DATE OR OUTAGE	ONE-TIME ACTION (Yes/No)	ON-GOING COMMITMENT (YES/NO)
1	Limitations regarding the operation with an inoperable LEFM system will be included in the LCS.	Prior to use above 3486 MWt	NO	YES
2	Necessary operating procedure revisions (including Emergency Operating Procedures and Abnormal Operating Procedures) will be completed prior to implementation of the proposed LEFM power uprate.	Prior to use above 3486 MWt	NO	YES
3	The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes.	Prior to use above 3486 MWt	YES	NO
4	Operator training will be completed prior to implementation of the proposed LEFM power uprate.	Prior to use above 3486 MWt	YES	NO
5	Plant testing for the proposed changes will be completed as described in Section 10.4, "Testing" of Enclosure 7	Prior to use above 3486 MWt	YES	NO
6	The plant process computer software will have a visual alarm at the Reactor Operator and Control Room Supervisor station displays to signal the operators to changes in the LEFM status.	Prior to use above 3486 MWt	YES	NO

Enclosure 8

Affidavits from GEH and the Electric Power Research Institute (EPRI) Supporting the Withholding of Information in Enclosure 7 from Public Disclosure

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-33853P, "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization," Revision 0, dated June 2016. GEH proprietary information in NEDC-33853P 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 9th day of June 2016.

James F. Harrison
Vice President, Fuel Licensing
GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Road
Wilmington, NC 28401
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ELECTRIC POWER RESEARCH INSTITUTE

NEIL WILMSHURST
Vice President and
Chief Nuclear Officer

Ref. EPRI Project Number 669

June 14, 2016

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:

Columbia Generating Station, Docket NO. 50-397 License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate, included in NEDC 33853P "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization", Class II (GEH Proprietary Information), June 2016

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 Energy Northwest. 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)

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

Columbia Generating Station, Docket NO. 50-397 License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate, included in NEDC 33853P "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization", Class II (GEH Proprietary Information), June 2016

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 a solid underline with highlighted text, inside double brackets. An example 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: 6-16-2016.

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 16th day of June, 2016, by Neil 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

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 9

GEH Report NEDO-33853, "Safety Analysis Report for Columbia Generating Station Thermal Power Optimization," Revision 0 (Non-Proprietary Version)

GE HITACHI

GE Hitachi Nuclear Energy

NEDO-33853
Revision 0
June 2016

Non-Proprietary Information- Class I (Public)

SAFETY ANALYSIS REPORT
FOR
COLUMBIA GENERATING STATION
THERMAL POWER OPTIMIZATION

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33853P, which has the proprietary information removed. Portions of the document that have been removed are indicated by a set of open and closed double square brackets as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

The design, engineering, and other information contained in this document are furnished for the purposes of supporting: a License Amendment Request by Energy Northwest, for a thermal power uprate at Columbia Generating Station to 3544 MWt in proceedings before the U.S. Nuclear Regulatory Commission. The only undertakings of GEH respecting information in this document are contained in the contract between GEH and Energy Northwest, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

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ACRONYMS AND ABBREVIATIONS

Term	Definition
ABA	Amplitude Based Algorithm
AC	Alternating Current
ACEC	Acute Critical Effluent Concentration
ADS	Automatic Depressurization System
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AP	Annulus Pressurization
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ART	Adjusted Reference Temperature
ASME	American Society Of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
B&PV	Boiler and Pressure Vessel
BHP	Brake Horsepower
BOP	Balance-of-Plant
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAC	Containment Atmosphere Control
CFD	Condensate Filter Demineralizer
CFR	Code of Federal Regulations
CGS	Columbia Generating Station
CIV	Containment Isolation Valve
CLTP	Current Licensed Thermal Power
CRD	Control Rod Drive
CRGT	Control Rod Guide Tube
CSC	Containment Spray Cooling

Term	Definition
CSS	Core Support Structure
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DC	Direct Current
DEH	Digital Electro-Hydraulic
DIVOM	Delta Critical Power Ratio Over Initial Minimum Critical Power Ratio Versus Oscillation Magnitude
ECCS	Emergency Core Cooling System
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
EOC	End-of-Cycle
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
FWTR	Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FPC	Fuel Pool Cooling
FSAR	Final Safety Analysis Report
FW	Feedwater
GDC	General Design Criteria
GE	General Electric Company
GEH	GE-Hitachi Nuclear Energy
GL	Generic Letter
GRA	Growth Rate Algorithm
GS3	GEH Simplified Stability Solution
HCOM	Hot Channel Oscillation Magnitude
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air

Term	Definition
HFCL	High Flow Control Line
HP	High Pressure
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilation and Air Conditioning
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICF	Increased Core Flow
ICH&GT	In-Core Housing and Guide Tube
IPE	Individual Plant Examination
IRM	Intermediate Range Monitor
JR	Jet Reaction
K1c	Stress Intensity for Crack Initiation
ksi	Kips Per Square Inch
kV	Kilovolt
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant-Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LTS	Long-Term Solution
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELC	Moderate Energy Line Crack
MELLLA	Maximum Extended Load Line Limit Analysis
MeV	Million Electron Volts
Mlb	Millions Of Pounds
MOV	Motor-Operated Valve
MS	Main Steam
MSF	Modified Shape Function

Term	Definition
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
MVA	Megavolt Amps
MWe	Megawatt(s)-Electric
MWt	Megawatt(s)-Thermal
NCL	Natural Circulation Line
NFI	New Fuel Introduction
NFWT	Nominal Feedwater Temperature
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
ODAF	Oil-Directed, Air-Forced
OFS	Orificed Fuel Support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
P/F	Power/Flow
P-T	Pressure-Temperature
PBDA	Period-Based Detection Algorithm
PCB	Polychlorinated Biphenyl
PCS	Pressure Control System
PCT	Peak Clad Temperature
PDC	Plant Design Change
PRA	Probabilistic Risk Assessment
PRFO	Pressure Regulator Failure Open – Maximum Steam Demand

Term	Definition
psi	Pounds Per Square Inch
psia	Pounds Per Square Inch – Absolute
psid	Pounds Per Square Inch – Differential
psig	Pounds Per Square Inch – Gauge
RBM	Rod Block Monitor
RCC	Reactor Building Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RFW	Reactor Feedwater
RFWP	Reactor Feedwater Pump
RFWT	Reduced Feedwater Temperature
RG	Regulatory Guide
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
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
SAR	Safety Analysis Report
SBO	Station Blackout
SC	Safety Communication
SCCS	Successive Confirmation Count Setpoint
SDC	Shutdown Cooling
SER	Safety Evaluation Report

Term	Definition
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SL	Safety Limit
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single-Loop Operating
SPC	Suppression Pool Cooling
SR	Surveillance Requirement
SRP	Standard Review Plan
SRSS	Square Root Sum of Squares
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
STP	Simulated Thermal Power
SW	Service Water
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TFSP	Turbine First-Stage Pressure
T/G	Turbine-Generator
TIP	Traversing In-Core Probe
TLO	Two (Recirculation) Loop Operation
TLTP	TPO Licensed Thermal Power
TLTR	NEDC-32938P-A, Thermal Power Optimization Licensing Topical Report
TPO	Thermal Power Optimization
TS	Technical Specification(s)
TSAR	Thermal Power Optimization Safety Analysis Report
TSV	Turbine Stop Valve
TSW	Plant Service Water
UHS	Ultimate Heat Sink
USE	Upper Shelf Energy
VWO	Valves Wide Open
Wd	Recirculation Drive Flow

EXECUTIVE SUMMARY

This report summarizes the results of all significant safety evaluations performed that justify increasing the licensed thermal power at Columbia Generating Station (CGS) to 3544 megawatts-thermal (MWt). The requested license power level is 1.66% above the current licensed thermal power (CLTP) level of 3486 MWt.

This report follows the Nuclear Regulatory Commission (NRC) approved format and content for boiling water reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32938P-A, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," called "TLTR." Per the outline of the TPO safety analysis report (TSAR) in the TLTR Appendix A, every safety issue that should be addressed in a plant-specific TPO licensing report is addressed in this report. For issues that have been evaluated generically, this report references the appropriate evaluation and establishes that the evaluation is applicable to the plant.

Only previously NRC approved or industry-accepted methods were used for the analysis of accidents, transients, and special events. Therefore, because the safety analysis methods have been previously addressed, they are not addressed in this report. Also, event and analysis descriptions that are provided in other licensing documents or the Final Safety Analysis Report (FSAR) 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%. A higher steam flow is achieved by increasing the reactor power along the current rod and core flow 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 CGS can safely operate at a power level of 3544 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 1.7% 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 FSAR 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 FSAR and/or applicable reload license.

Any modification to safety-related or non-safety-related equipment will be implemented in accordance with 10 Code of Federal Regulations (CFR) 50.59.

All plant systems and components 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 assure that the increased thermal power level continues to comply with the existing plant environmental regulations.

An assessment, as defined in 10 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 FSAR and approved design changes ensures adequate evaluation of the licensing basis for the effect of TPO through the date of that evaluation.

The plant licensing requirements have been reviewed, and it is concluded that this TPO can be accommodated (1) without a significant increase in the probability or consequences of an accident previously evaluated, (2) without creating the possibility of a new or different kind of accident from any accident previously evaluated, and (3) 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 document addresses a thermal power optimization (TPO) power uprate of 1.66% of the current licensed thermal power (CLTP), consistent with the magnitude of the thermal power uncertainty reduction for the Columbia Generating Station (CGS) plant. This will result in an increase in licensed thermal power from 3486 MWt to 3544 MWt and an increase in electrical power from 1206 megawatts-electric (MWe) to 1227 MWe.

This report follows the Nuclear Regulatory Commission (NRC)-approved format and content for boiling water reactor (BWR) TPO licensing reports 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 provided in NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) (Reference 2) and NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR2) (Reference 3). Since their NRC approval, numerous extended power uprate (EPU) submittals have been based on these reports. The outline for the TPO safety analysis report (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 are included in this TSAR. For issues that have been evaluated generically, this report references the appropriate evaluation and establishes that it is applicable to CGS.

BWR plants, as currently licensed, have safety systems and component capability for operation at least 1.5% above the CLTP level. 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%. Several Pressurized Water Reactor and BWR plants have already been authorized 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, the ≥ 102% safety analysis basis is reestablished above the uprated power level. Therefore, all GEH BWR plant designs have the capability to implement a TPO uprate, whether or not the plant has previously been uprated.

1.2 PURPOSE AND APPROACH

1.2.1 TPO Analysis Basis

CGS was originally licensed at 3323 MWt. CGS was uprated to the CLTP level of 3486 MWt through the issuance of Amendment 137 to the facility operating license. The current safety analysis basis assumes, where required, 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 remain applicable at the TPO rated thermal power (RTP), because the 2% factor from Regulatory Guide (RG) 1.49, "Power Levels of Nuclear Power Plants," is effectively reduced by the improvement in the FW flow measurements. Some analyses may be performed at TPO RTP, because the uncertainty factor is accounted for in the methods, or the additional 2% margin is not required (e.g., anticipated transient without scram (ATWS)). Detailed descriptions of the basis for the TPO analyses are provided in the subsequent sections of this report.

Figure 1-1 illustrates the TPO power/flow (P/F) operating map for the analysis at 101.66% of CLTP for CGS. The changes to the P/F operating map are consistent with the generic descriptions given in TLTR Section 5.2. The approach to achieve a higher thermal power level is to increase core flow along the established Maximum Extended Load Line Limit Analysis (MELLLA) rod line. This strategy allows CGS to maintain most of the existing available core flow operational flexibility while assuring that low power-related issues (e.g., stability and ATWS instability) do not change because of the TPO uprate.

No increase in the previously licensed maximum core flow limit is associated with the TPO uprate. 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 operation (SLO) operating domain as a result of the TPO uprate. Therefore, the SLO operating domain is not provided. For CGS, the maximum analyzed reactor core thermal power for SLO remains at the licensed limit.

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, etc. 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 TPO uprate does not affect the pressure control function of the turbine bypass valves.

1.2.2 Margins

The TPO analysis basis ensures that the power-dependent instrument error margin identified in RG 1.49 is maintained. 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 1-1. Computer codes used in previous analyses (i.e., analyses at 102% of CLTP) are not listed. Similarly, 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.

1.2.3 Scope of Evaluations

The scope of evaluations is discussed in TLTR Appendix B. Tables B-1 through B-3 identifies those analyses that are bounded by current analyses, those that are not significantly affected, and those that require updating. The disposition of the evaluations as defined by Tables B-1 through

B-3 is applicable to CGS. This TSAR includes all of the evaluations for the plant-specific application. Many of the evaluations are supported by generic reference, some supported by rational considerations of the process differences, and some plant-specific analyses are provided.

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 confirmed for each fuel cycle, and will continue to be evaluated and documented for each fuel cycle.

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) 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 the cycle of interest.

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 4).

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.

1.2.5 Concurrent Changes Unrelated to TPO

No concurrent changes unrelated to TPO are included in this evaluation because there are no other pending license amendments.

1.3 TPO PLANT OPERATING CONDITIONS

1.3.1 Reactor Heat Balance

The typical heat balance diagrams at the TPO conditions are presented in Figure 1-2 (Reactor Heat Balance – TPO Power at 101.66% of CLTP, 100% Core Flow).

The small changes in thermal-hydraulic parameters for the 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 CGS operation is considered to match expected TPO uprate conditions.

1.3.2 Reactor Performance Improvement Features

The following performance improvement and equipment out-of-service (OOS) features currently licensed at CGS are acceptable at the TPO RTP level:

Performance Improvement Feature
Single Loop Operation
Increased Core Flow (ICF) (106.0% of rated)
MELLLA (82.7% of Rated Core Flow at TPO Licensed Thermal Power (TLTP))
Feedwater Temperature Reduction (FWTR), 355°F
Feedwater Heater(s) OOS, 355°F
SRV OOS (12 Valves in Service) / Automatic Depressurization System (ADS) 2 Valves OOS
Turbine Bypass Valve (TBV) OOS
Recirculation Pump Trip (RPT) OOS

1.4 Basis for TPO Uprate

The safety analyses in this report are based on a total thermal power measurement uncertainty of 0.3%. This will bound the actual power level requested. The detailed basis value is provided in CGS plant design change (PDC) EC-14942, which addresses the improved FW flow measurement accuracy using the Caldon Leading Edge Flow Meter Check-Plus system.

1.5 Summary and Conclusions

This evaluation has investigated a TPO uprate to 101.66% of CLTP. The strategy for achieving this higher power is to increase core flow along the established MELLLA rod lines. The plant licensing challenges have been reviewed (Table 1-3) to demonstrate how the TPO uprate 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 or design allowable limits applicable to the plant which might cause a reduction in a margin of safety. The TPO uprate described herein involves no significant hazards consideration.

Table 1-1 Computer Codes For TPO Analyses

Task	Computer Code	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y(1)	NEDE-24011P Rev. 0 SER
Thermal-Hydraulic Stability	ODYSY ISCOR PANAC TRACG	05 09 11 04	Y Y(1) Y(2) Y	NEDE-33213P-A NEDE-24011P Rev. 0 SER NEDE-30130-A NEDE-33147P-A Rev. 4
Reactor Internal Pressure Differences	ISCOR	09	Y(1)	NEDE-24011P Rev. 0 SER
Piping Components Flow Induced Vibration (FIV)	SAP4G07	07	N(3)	NEDO-10909

* The application of these codes to the CGS TPO analyses complies with the limitations, restrictions, and conditions specified in the approving NRC Safety Evaluation Report (SER) where applicable for each code.

Notes for Table 1-1:

(1) The ISCOR code is not approved by name. However, the SER supporting approval of NEDE-24011P Revision 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) 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.

(2) The use of 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 Licensing Topical Report NEDE-24011P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(3) 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. Also, the application of this code has been used in previous power uprate submittals.

Table 1-2 Thermal-Hydraulic Parameters at TPO Uprate Conditions

Parameter	**CLTP**	**TPO RTP (101.66% of CLTP)**
Thermal Power (MWt) (Percent of Current Licensed Power)	3,486 100.0	3,544 101.66
Steam Flow (Mlb/hr) (Percent of Current Rated)	15.016 100.0	15.284 101.8
FW Flow (Mlb/hr) (Percent of Current Rated)	14.985 100.0	15.253 101.8
Dome Pressure (psia)	1,035	1,035
Dome Temperature (°F)	548.8	548.8
FW Temperature (°F)	421.2	422.1
Full Power Core Flow Range (Mlb/hr) (Percent of Current Rated)	87.6 to 115.0 (80.7 to 106.0)	89.7 to 115.0 (82.7 to 106.0)

Table 1-3 Summary of Effect of TPO Uprate on Licensing Criteria

Key Licensing Criteria	Effect of 1.66% Thermal Power Increase	Explanation of Effect
LOCA challenges to fuel (10 CFR 50, Appendix K)	No increase in peak clad temperature (PCT), no change of maximum LHGR required.	Previous analysis accounted for ≥ 102% of licensed power, bounding TPO operation. No vessel pressure increase.
Change of operating limit MCPR (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 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	Approximately 1.66% increase. Must remain within 10 CFR 20 limits.	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.	No change in harsh environment terms (TPO operating conditions bounded by previous analyses); minimal change in normal operating conditions.
Fracture toughness, 10 CFR 50, Appendix G	< 2°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-core flow 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 (30 psig), must stay within existing American Society of Mechanical Engineers (ASME) Code "Emergency" category stress limit.	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.

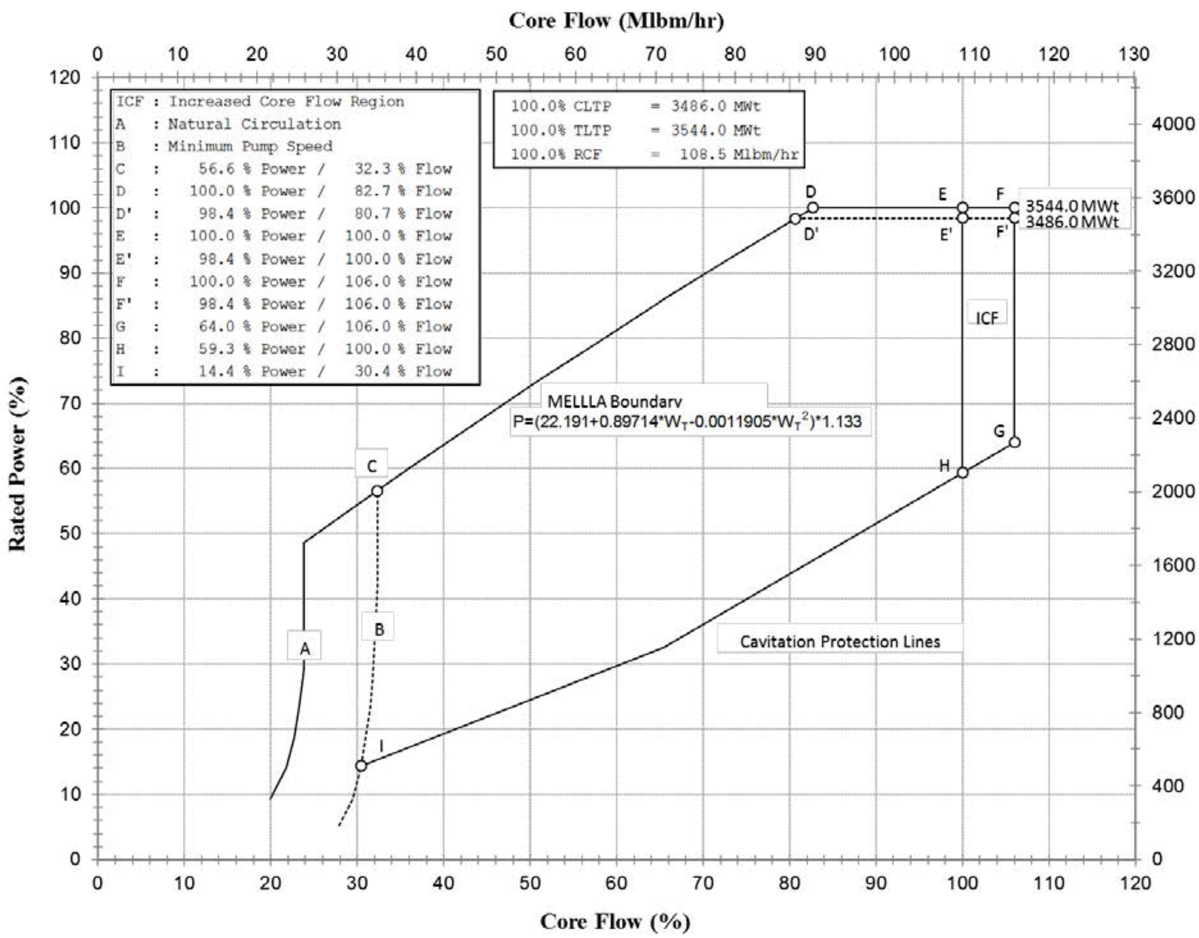
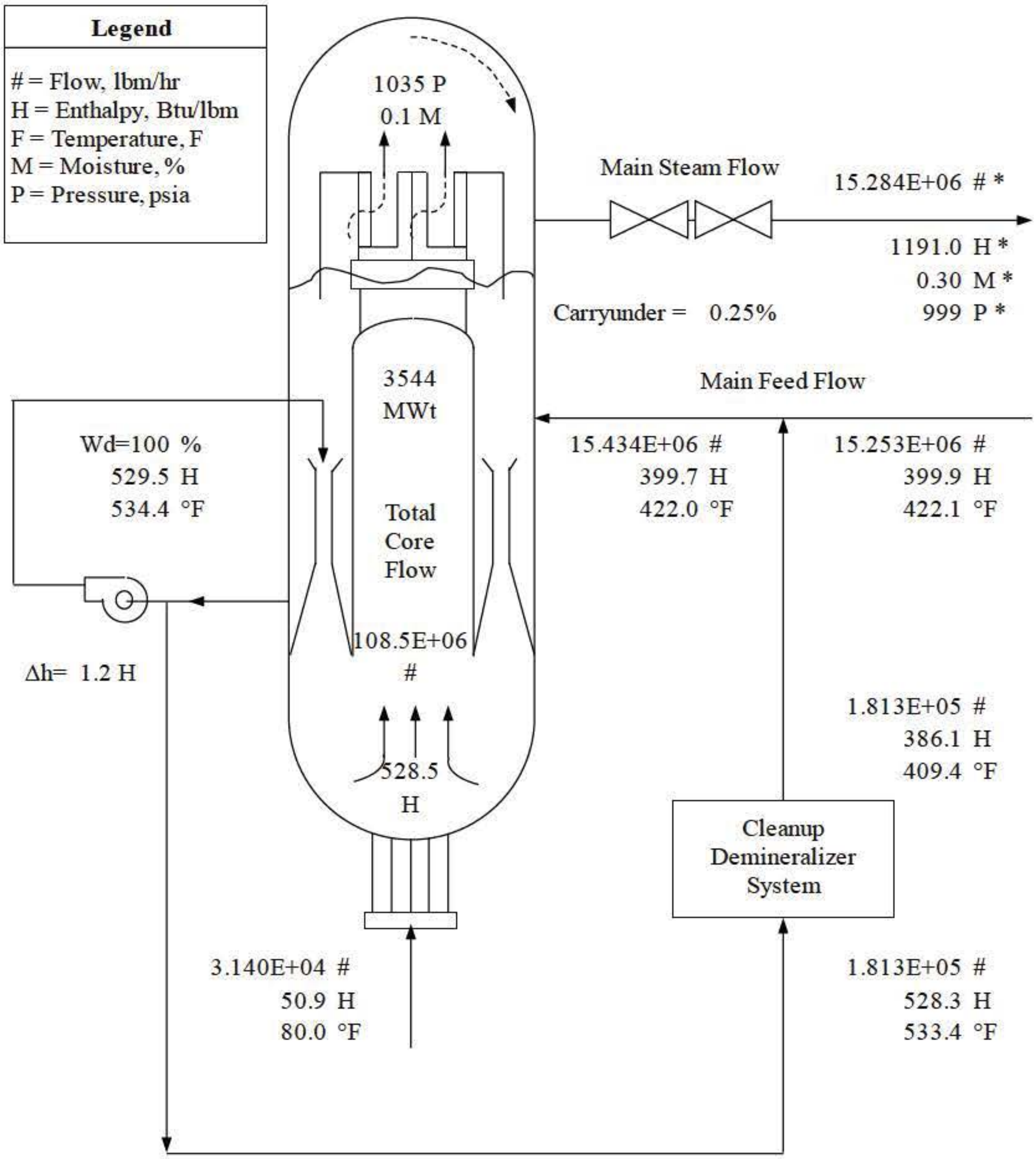


Figure 1-1 Power/Flow Map for TPO (101.66% of CLTP)



*Conditions at upstream side of TSV

Core Thermal Power	3544.0
Pump Heating	11.5
Cleanup Losses	-7.6
Other System Losses	-1.1
Turbine Cycle Use	3546.8 MWt

Figure 1-2 Reactor Heat Balance – TPO Power (101.66% of CLTP), 100% Core Flow

2.0 REACTOR CORE AND FUEL PERFORMANCE

2.1 FUEL DESIGN AND OPERATION

At the TPO RTP conditions, all fuel and core design limits are met by the deployment of fuel enrichment and burnable poison, control rod pattern management, and core flow adjustments. New fuel designs are not needed for the TPO to ensure safety. However, 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 fuel design is adequate for TPO operation.

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, loss-of-coolant-accident (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.

The historical 25% of RTP value for the TS SL, some thermal limits monitoring Limiting Condition for Operation (LCO) thresholds, and some Surveillance Requirement (SR) thresholds are based on [[

]] The historical 25% RTP value is a conservative basis, as described in the plant TS; [[

]] Therefore, the SL percent RTP basis, some thermal limits monitoring LCOs, and SR percent RTP thresholds remain at 25% RTP for the TPO uprate.

2.2.1 Safety Limit MCPR

The safety limit minimum critical power ratio (SLMCPR) is dependent upon the nominal average power level and the uncertainty in its measurement. Consistent with approved practice, a revised SLMCPR is calculated for the first TPO fuel cycle and confirmed for each subsequent cycle. The historical uncertainty allowance and calculational methods are discussed in TLTR Section 5.7.2.1.

2.2.2 MCPR Operating Limit

TLTR Appendix E shows that the changes in the operating limit minimum critical power ratio (OLMCPR) for a TPO uprate [[

]] 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.

2.2.3 MAPLHGR and Maximum LHGR Operating Limits

The maximum average planar linear heat generation rate (MAPLHGR) and maximum linear heat generation rate (LHGR) limits are maintained as described in TLTR Section 5.7.2.2. No significant change results due to TPO operation. The LHGR limits are fuel dependent and are not affected by the TPO. The ECCS performance is addressed in Section 4.3.

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.

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. Through fuel cycle redesign, sufficient excess reactivity can be obtained to match the desired cycle length.

2.4 THERMAL HYDRAULIC STABILITY

CGS is operating under the requirements of reactor stability Long-Term Solution Option III. The Option III solution monitors Oscillation Power Range Monitor (OPRM) signals to determine when a reactor scram is required. The OPRM signal is evaluated by the Option III stability algorithms to determine when the signal is becoming sufficiently periodic and large to warrant a reactor scram to disrupt the oscillation (Reference 5). The OPRM system may only cause a scram when plant operation is in the Option III Armed Region. For TPO operation, the Armed Region is modified to maintain the CLTP absolute power of 872 MWt (24.6% of the planned TPO uprated power of 3544 MWt) and flow (60% of rated recirculation drive flow). The stability based OLMCPR associated with the OPRM setpoint assures that the Critical Power Ratio (CPR) SL is not violated following an instability event. This is validated for every reload cycle.

2.4.1 Stability Option III

CGS has implemented the stability long-term solution (LTS) Option III (References 5 and 6). The Option III solution combines closely spaced local power range monitor (LPRM) detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed oscillation power range monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The period-based detection algorithm (PBDA) is the only algorithm credited in the Option III licensing basis (Reference 5). Two defense-in-depth algorithms, referred to as the amplitude based algorithm (ABA) and the growth rate algorithm (GRA), offer a higher degree of assurance that fuel failure will not occur as a consequence of stability-related oscillations. Because the OPRM hardware does not change, the hot channel oscillation magnitude (HCOM) portion of the Option III calculation is not affected by TPO and does not need to be recalculated.

The Option III OPRM Trip-Enabled Region has been defined as the region where the OPRM system is fully armed (≤ 60% rated core flow and ≥ 25.0% CLTP). For TPO, the Option III OPRM Trip-Enabled Region is rescaled to maintain the same absolute P/F region boundaries. The backup stability protection (BSP) evaluation described in Section 2.4.2 shows that the generic Option III OPRM Trip-Enabled Region is adequate. The OPRM Trip-Enabled Region is demonstrated in Figure 2-1.

Because the rated core flow does not change for TPO, the 60% core flow boundary is not rescaled. The 25.0% CLTP boundary changes by the following equation:

$$\text{TPO Region Boundary} = 25.0\%\ \text{CLTP} \div 100\%\ \text{TPO}\ (\%\ \text{CLTP})$$

Thus, for a 101.66% of CLTP TPO:

$$\text{TPO Region Boundary} = 25.0\%\ \text{CLTP} \div 101.66\%\ \text{CLTP} = 24.6\%\ \text{TPO}$$

The minimum power level at which the OPRM should be confirmed operable is 19.6% TPO. A 5% absolute power separation between the OPRM Trip-Enabled Region power boundary and the power at which the OPRM system should be confirmed operable is deemed adequate for the Option III application.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability that exceeds the specified trip setpoint is detected. The demonstration setpoint is determined per the current NRC approved methodology. The Option III stability reload licensing basis calculates the OLMCPRs required to protect the SLMCPR for both steady-state and transient stability events as specified in the Option III methodology (Reference 6). These OLMCPRs are calculated for a range of OPRM amplitude setpoints for TPO operation. Selection of an appropriate instrument setpoint is then based upon the OLMCPR required to provide adequate SLMCPR protection. This determination relies on the delta critical power ratio over initial minimum critical power ratio versus oscillation magnitude (DIVOM) to determine an OPRM amplitude setpoint that protects the SLMCPR during an anticipated instability event. A DIVOM analysis is performed and used in the Option III OPRM amplitude setpoint demonstration.

As demonstrated in Table 2-1, with an estimated OLMCPR of 1.45 and an estimated SLMCPR of 1.10, an OPRM amplitude setpoint of 1.13 with an OPRM successive confirmation count setpoint (SCCS) of 15 (Reference 6) is the highest setpoint that may be used without stability setting the OLMCPR. The actual setpoint will be established in accordance with CGS TS at each reload. These demonstration results are based on a power level of 101.66% CLTP.

The NRC-approved GEH Simplified Stability Solution (GS3) also applies to CGS at TPO conditions. The trip setpoint can be determined per GS3 methodology (Reference 7) based on BWR/5 plants with Option III.

Therefore, TPO operation is justified for plant operation with stability LTS Option III.

2.4.2 Stability Backup Stability Protection

CGS has implemented the BSP methodology (Reference 8) as the stability backup solution should the OPRM system be declared inoperable.

The BSP regions consist of two regions, I-Scram and II-Controlled Entry. The Base BSP Scram Region and the Base BSP Controlled Entry Region are defined by state points on the high flow control line (HFCL) and on the natural circulation line (NCL) in accordance with Reference 8. The bounding plant-specific BSP region state points must enclose the corresponding base BSP region state points on the HFCL and on the NCL. If a calculated BSP region state point is located inside the corresponding base BSP region state point, then it must be replaced by the corresponding base BSP region state point. If a calculated BSP region state point is located outside the corresponding base BSP region state point, this point is acceptable for use. That is, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries are constructed by connecting the corresponding bounding state points on the HFCL and the NCL using a shape function. The modified shape function (MSF, Reference 9) is applied to this analysis.

The demonstration BSP regions for both the nominal feedwater temperature (NFWT) and the reduced feedwater temperature (RFWT) operations are shown in Table 2-2 and Figure 2-2, and Table 2-3 and Figure 2-3, respectively. The OPRM Trip-Enabled Region is confirmed for NFWT and RFWT operations based on the demonstration BSP regions for NFWT and RFWT. These demonstration results are based on a power level of 101.66% CLTP.

The BSP regions are confirmed or expanded on a cycle-specific basis.

Therefore, TPO operation is justified for plant operation with stability BSP regions.

2.5 REACTIVITY CONTROL

The generic discussion in TLTR Section 5.6.3 and Appendix J.2.3.3 applies to CGS. The control rod drive (CRD) and CRD hydraulic systems and supporting equipment are not affected by the TPO uprate and no further evaluation of CRD performance is necessary.

Table 2-1 OPRM Amplitude Setpoint Versus OLMCPR Demonstration

OPRM Amplitude Setpoint	CGS TPO	
	OLMCPR(2RPT)	OLMCPR(SS)
1.05	1.27	1.19
1.06	1.29	1.21
1.07	1.31	1.23
1.08	1.33	1.25
1.09	1.35	1.27
1.10	1.37	1.29
1.11	1.40	1.31
1.12	1.42	1.33
1.13	1.44	1.35
1.14	1.47	1.37
1.15	1.49	1.40
Acceptance Criteria	**Rated Power OLMCPR**	**Off-Rated OLMCPR at 45% Flow**

Table 2-2 BSP Region Intercepts for Nominal Feedwater Temperature Demonstration

Region Boundary Intercept	% TPO Power	% Core Flow
Region (Region I) Boundary Intercept on HFCL Scram		
A1-Base	63.6	40.0
Scram Region (Region I) Boundary Intercept on NCL		
B1	37.1	23.8
Controlled Entry Region (Region II) Boundary Intercept on HFCL		
A2-Base	72.6	50.0
Controlled Entry Region (Region II) Boundary Intercept on NCL		
B2 Base	28.1	23.7

Table 2-3 BSP Region Intercepts for Reduced Feedwater Temperature Demonstration

Region Boundary Intercept	% TPO Power	% Core Flow
Scram Region (Region I) Boundary Intercept on HFCL		
A1	64.8	41.3
Scram Region (Region I) Boundary Intercept on NCL		
B1	32.0	23.8
Controlled Entry Region (Region II) Boundary Intercept on HFCL		
A2-Base	72.6	50.0
Controlled Entry Region (Region II) Boundary Intercept on NCL		
B2-Base	28.1	23.7

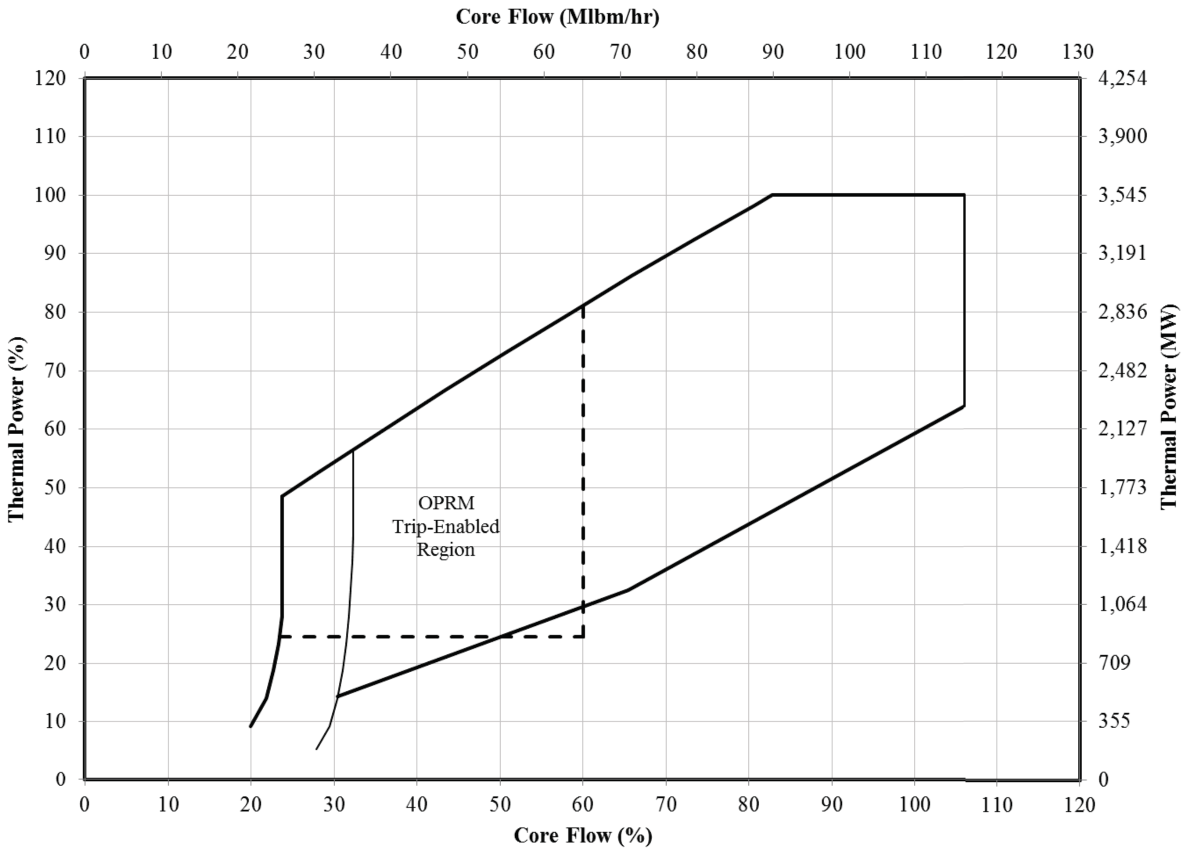


Figure 2-1 Illustration of OPRM Trip-Enabled Region

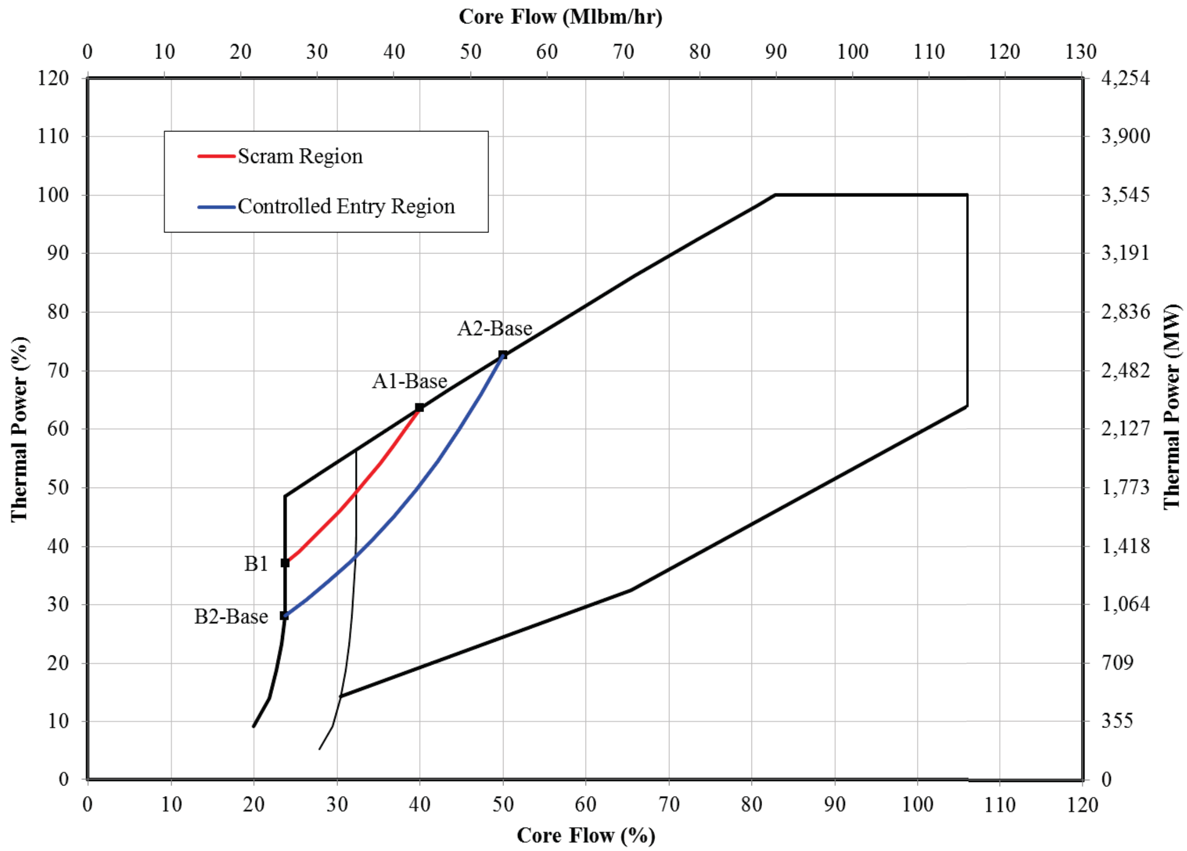


Figure 2-2 Demonstration BSP Regions for Nominal Feedwater Temperature

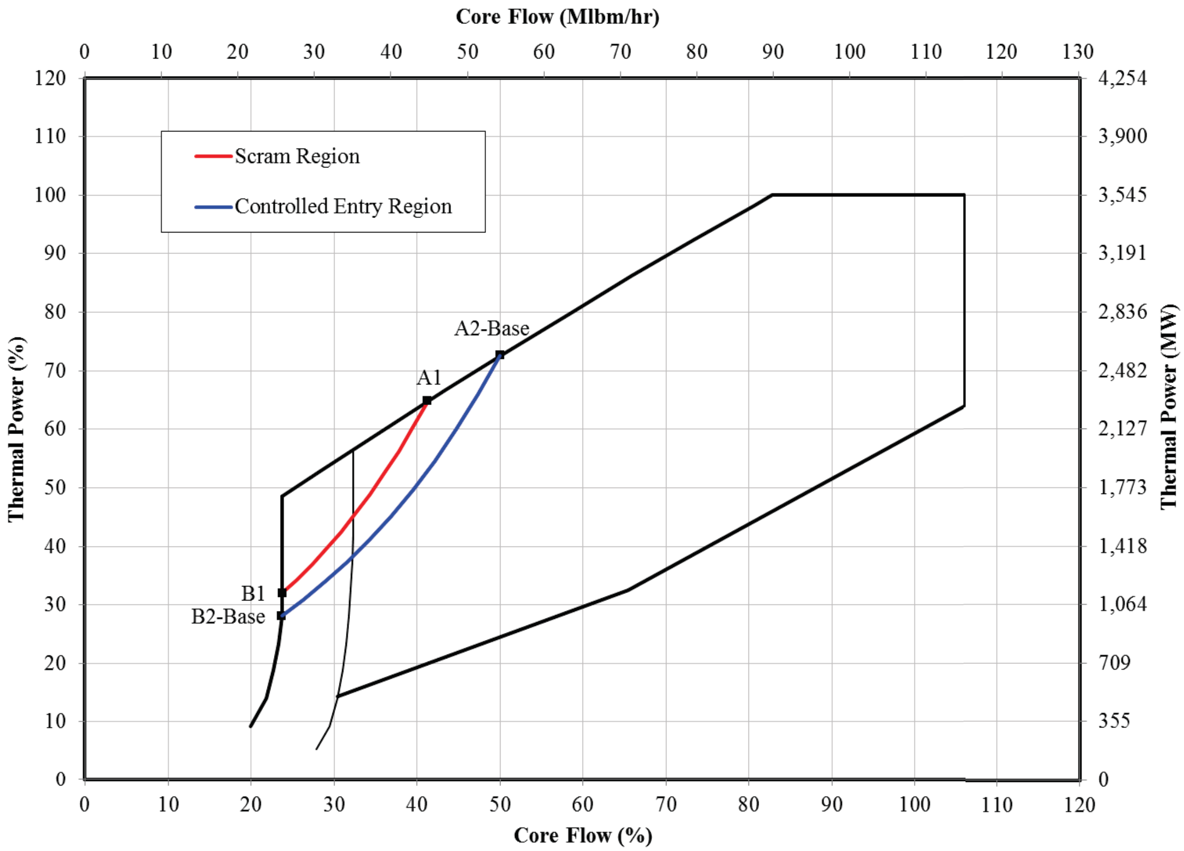


Figure 2-3 Demonstration BSP Regions for Reduced Feedwater Temperature

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. Evaluations and analyses for the CLTP have been 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. There is no increase in nominal operating pressure for the CGS TPO uprate. There are no changes in the SRV setpoints or valve OOS options. There is no change in the methodology or the limiting overpressure event. Therefore, the generic evaluation contained in the TLTR is applicable.

The analysis for each fuel reload, which is current practice, confirms the capability of the system to meet the ASME design criteria.

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. CGS is evaluated for a fluence that bounds the required value for operation at TPO conditions.

The neutron fluence for TPO is calculated using two-dimensional neutron transport theory. The neutron transport methodology is consistent with RG 1.190. A bounding peak fluence 1.15E+18 n/cm^2 is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G (Reference 10). 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 maintain the margin requirements of 10 CFR 50, Appendix G as defined in RG 1.99 (Reference 11). Many of the CGS RPV materials do not have sufficient unirradiated USE data. Therefore, equivalent margin analyses were performed for the limiting beltline plate, weld, and nozzle forging materials to assure qualification. These values are provided in Tables 3-1, 3-2 and 3-3 for CGS.

(b) The beltline material reference temperature of the nil-ductility transition (RT_{NDT}) remains below the 200°F screening criteria as defined in Reference 11. These values are provided in Table 3-4 for CGS.

(c) The TPO end of cycle 27 cumulative energy is less than the CLTP cumulative energy, therefore the CLTP PT curves remain bounding for TPO, limited to the currently approved fluence / EFPY (33.10 at OLTP Power). The current Adjusted Reference Temperature (ART) values for the beltline plates and welds from the PT curve report increase slightly for TPO conditions. However they remain bounded for TPO when compared to the TLAA evaluation (considers an EFPY and resulting higher fluence level). The currently licensed PT curves include the Low Pressure Coolant Injection (LPCI) nozzle. The water level instrumentation nozzle that occurs within the beltline region is bounded by the CLTP curves.

(d) The surveillance program consists of three capsules in CGS. One capsule (at the 300 degree azimuthal location) was removed in the Spring 1996 (after 7.2 EFPY of operation) and tested. A reconstituted 300 degree azimuth capsule was re-installed in May 1997. The 120 degree azimuth capsule holder fell off prior to the 1989 in-vessel examination and was re-installed during the 1991 outage. The one remaining capsule (30 degree azimuthal) has been in the reactor vessel since plant startup. CGS is a participant in the Integrated Surveillance Program, currently administrated by EPRI, and is not designated as a host plant; therefore, no capsules are slated for removal at this time. TPO has no effect on the existing surveillance schedule.

(e) The 51.56 EFPY beltline axial and circumferential weld material RT_{NDT} remains bounded by the requirements of Boiling Water Reactor Vessel and Internals Project (BWRVIP)-05 as defined in References 15 and 16. This comparison is provided in Tables 3-5 and 3-6 for axial and circumferential welds, respectively.

(f) An evaluation on brittle fracture of the RPV due to reflood following a postulated loss-of-coolant-accident 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 loss-of-coolant-accident is evaluated for the maximum ART value for TPO. The evaluation calculates the temperature required to achieve a fracture toughness of 200 ksi-in^0.5 when using the equation for fracture toughness stress intensity for crack initiation (K1c) presented in Appendix A of ASME Section XI. This calculated temperature (196.25 °F) is well below the minimum 400 °F temperature predicted for the thermal shock event at the time of Peak stress intensity.

(g) 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 loss-of-coolant-accident is evaluated for the maximum ART value for TPO. The evaluation calculates the temperature required to achieve a fracture toughness of 200 ksi-in^0.5 when using the equation for fracture toughness K1c presented in Appendix A of ASME Section XI. This calculated temperature (196.25 °F) is 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 with TPO does not have an adverse effect (not exceeding regulatory requirements) on the reactor vessel fracture toughness.

3.2.2 Reactor Vessel Structural Evaluation

[[

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The TLTR provides a generic disposition for components that are not significantly affected. The following table provides the justification for confirming the TLTR generic disposition:

Topic	TLTR Generic Parameter(s) or Requirement(s)	Justification / CLTP vs. 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.

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 1971 Edition with addenda to and including the Summer 1971 Addenda was used as the governing code and Paragraph NB-3338.2 of the Winter 1971 Addenda shall supersede Paragraph NB-3338.2 of the 1971 Edition. 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 scaling the "original" stresses based on the TPO conditions (pressure, temperature, and flow). The bounding analyses were performed for the design, normal and upset, and emergency and faulted conditions. If there is an increase in annulus pressurization (AP), jet reaction (JR), pipe restraint or fuel lift loads, the changes 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 Code requirements are met.

3.2.2.2 Normal and Upset Conditions

The reactor coolant temperature and flows at TPO conditions are unchanged from those at current rated conditions, because the 105% 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. A primary plus secondary stress analysis was performed showing TPO stresses still meet the requirements of the ASME Code, Section III, and Subsection NB for all components. The CGS fatigue analysis results for the limiting components are provided in Table 3-7. The CGS analysis results for TPO 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.

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 Difference

The reactor internal pressure differences (RIPDs) are affected more by the maximum licensed core flow rate than by the power level. The maximum licensed core flow rate is not changed 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 bounded by the current analysis that conservatively assumed an initial power level of 110% OLTP (104.1% CLTP) for Normal and 112% OLTP (106.2% CLTP) for Upset. The Emergency and Faulted evaluations of RIPDs for the TPO uprate are bounded by the current analyses that conservatively assumed an initial power level of 112% of OLTP (106.2% CLTP).

Fuel bundle lift margins and control rod guide tube (CRGT) lift forces are calculated at the Faulted condition to demonstrate that fuel bundles would not lift under the worst conditions. The current analysis conservatively assumed 112% OLTP (106.2% CLTP) and 106% core flow, which bounds the TPO. The effect due to the changes in minimum fuel lift margins and maximum CRGT lift forces is reported in Section 3.3.2.

Acoustic and flow-induced loads on jet pump, core shroud and shroud support due to a recirculation line break (RLB) are bounded by the current analyses that are calculated based on Safety Communication (SC) 12-20 (Reference 13).

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 Support
- Shroud
- Core Plate
- Top Guide
- Control Rod Drive Housing and Mechanism
- Control Rod Guide Tube
- Orificed Fuel Support (OFS)

Non-CSS Components

- FW Sparger

- Jet Pump Assembly
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separator Assembly
- In-Core Housing and Guide Tube
- Core Differential Pressure and Liquid Control Line
- Low Pressure Coolant Injection Coupling

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, SRV, LOCA, AP/JR, acoustic and flow induced loads due to RLB, fuel lift, hydraulic flow and thermal loads.

RPV design pressure remains unchanged. RIPD loads are bounded by GE14 New Fuel Introduction (NFI) values (111% of OLTP and 106% ICF). Seismic, SRV, LOCA and AP/JR loads remain unchanged. Acoustic and flow induced loads due to RLB remain bounded for TPO, but increase due to GEH SCs. The increase in hydraulic flow and thermal load is insignificant. 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.

GEH SCs SC 12-20 (Reference 13), SC 14-02 (Reference 22), and SC 14-03 (Reference 23) were evaluated and resulted in an acoustic load increase for some RPV internals. The stresses of the RPV internals that were affected by the SCs were reconciled for the increase of the acoustic load to show that adequate stress margins still exist and the stresses remain within the allowable limits. All the RPV internals were shown to be within the allowable limits. The limiting stresses of all RPV internal components are summarized in Table 3-8. Therefore the RPV internal components are demonstrated to be structurally qualified for operation at TPO conditions.

3.3.3 Steam Separator and Dryer Performance

For CGS, the TPO performance of the steam dryer/separator was evaluated. The results of the evaluation demonstrated that the steam dryer/separator 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 core flow, this results in an increase in the separator inlet quality, an increase in the steam dryer face velocity and a decrease in the water level inside the dryer skirt. These factors, in addition to the radial power distribution, affect the steam dryer/separator performance. However, the net effect of these changes does not result in exceeding the acceptable moisture content of ≤ 0.10 wt. % leaving the steam dryer. In addition, the changes in separator and dryer performance do not result in unacceptable water levels inside the dryer skirt.

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 flow-induced vibration (FIV) on the reactor internals at 106% rated core flow and TPO RTP of 101.66% of CLTP. The vibration levels for the TPO conditions were estimated from measured vibration data during startup tests on CGS and the NRC designated prototype plant (Tokai-2), as well as other plants. The 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
FW Sparger	FW flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV. Extrapolation of measured data shows stresses are within limits.
Jet Pumps	The increase in jet pump flow at TPO is negligible based on no change in core flow and a minor increase in core differential pressure (< 0.1 psi).	Slight increase (< 2%) in FIV. Extrapolation of measured data shows stresses are within limits.
Jet Pump Sensing Lines	Resonance at vane passing frequency	No resonance at vane passing frequency range due to TPO. Clamps have been installed at all JPSLs to prevent resonance with VPF.
Shroud	Flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV. The maximum stresses are well within limits.
Shroud Head and Separator	Steam flow at TPO RTP is approximately 2% greater than CLTP.	Slight increase (< 4%) in FIV. Extrapolation of measured data shows stresses are within limits.
CRGT and In-Core Guide Tubes	Core flow at TPO is unchanged from CLTP.	No change.

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 are 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.

Therefore, it is concluded that the flow-induced vibrations for all evaluated components remain within acceptable limits.

The safety-related main steam (MS) piping has minor increased flow rates and flow velocities resulting from the TPO uprate. CGS has no safety-related thermowells and sample probes installed in the FW system.

The piping components were evaluated in accordance with ASME Code N-1300 (Reference 12) FIV analysis guidelines. The resonance separation rule in ASME Appendix N Subparagraph N-1324.1(d) of Reference 12 was used to determine if adequate separation exists between the vortex shedding frequencies and the natural frequencies of the piping components.

The MS piping experiences increased vibration levels, approximately proportional to the increase in the square of the flow velocities and also in proportion to any increase in fluid density. The MS piping vibration is expected to increase only by about 4% from 3.753 million pounds (Mlb)/hr per line at CLTP to 3.822 Mlb/hr per line at TPO. 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. Therefore, the MS lines vibration will remain within acceptable limits during TPO. Analytical evaluation has shown that the safety-related thermowells and sample probes in the MS and recirculation piping systems are structurally adequate for the TPO operating conditions.

3.5 Piping Evaluation

3.5.1 Reactor Coolant Pressure Boundary Piping

The methods used for the piping and pipe support evaluations are described in TLTR Appendix K. These approaches are identical to those used in the evaluation of previous BWR power uprates of up to 20% power. 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 increases ~1°F.	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 (SRVDL) piping up to first anchor, reactor core isolation cooling (RCIC) MS drain lines, RPV head vent line piping located inside containment) Pipe Stresses Pipe Supports Flow-Accelerated Erosion/Corrosion (FAC)	Nominal dome pressure at TPO RTP is identical to CLTP. Steam flow at TPO RTP is ~ 2% greater than CLTP. Minor decrease in main steam line (MSL) pressure < 2 psi.	Plant specific evaluation performed Minor change in pipe stress Minor effect on pipe supports Minor increase in the potential for FAC (FAC concerns are covered by existing piping monitoring program)
FW and Attached Piping (Inside Containment) Pipe Stresses Pipe Supports FAC	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. Fluid temperature remains the same.	Plant specific evaluation performed Negligible change in pipe stress Negligible effect on pipe supports Minor increase in the potential for FAC (FAC concerns are covered by existing piping monitoring program)

Component(s) / Concern	Process Parameter(s)	TPO Evaluation
RPV Bottom Head Drain Line, RCIC Piping, High Pressure Core Spray (HPCS) Piping, LPCI Piping, LPCS Piping, SLCS Piping, and Reactor Water Cleanup (RWCU) Piping Pipe Stresses Pipe Supports FAC	Nominal dome pressure at TPO RTP is identical to CLTP. Small increase in core pressure drop of < 1 psi. Recirculation fluid temperature increases ~1°F.	Negligible change in pipe stress Negligible effect on pipe supports Minor increase in the potential for FAC (FAC concerns are covered by existing piping monitoring program)

For the MS and FW lines, supports, and connected lines, the methodologies as described in TLTR Section 5.5.2 and Appendix K were used to determine the percent increases in applicable ASME Code stresses, displacements, 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 load, 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 at the TPO uprate conditions.

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. CGS has an established FAC monitoring program 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 method to calculate wall thinning of components susceptible to FAC. For TPO, the evaluation of predicted wall thinning of the MS and attached piping indicates minimal effect.

No changes to piping inspection scope and frequency are required prior to TPO implementation to ensure adequate margin for the changing process conditions. The continuing inspection program will take into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance any adverse effect from TPO on high-energy piping systems potentially susceptible to pipe wall thinning due to FAC is monitored and addressed.

FW Piping System Evaluation

The FW piping system (inside containment) was evaluated for compliance with the ASME Section III 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 can be affected by FAC. FAC in the FW piping is affected by changes in fluid velocity and temperature. CGS has an established program for monitoring pipe wall thinning in single 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 includes the use of a predictive method to calculate wall thinning of components susceptible to FAC. For TPO, the evaluation of predicted wall thinning of the FW Piping System indicates minimal effect.

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 continuing inspection program will take into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance 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.

The piping systems evaluated are as follows:

(1) Main Steam (MS) (outside containment) including equalization header, turbine bypass piping, and crossover piping
(2) Bleed Steam (BS)
(3) Seal Steam (SS)
(4) Condensate Storage and Transfer (CST)
(5) Heater/ MSR Vents and Drains (HD/HV)
(6) Reactor FW (RFW) (outside containment)
(7) Condensate (COND)

The following piping systems have no change in operating conditions between CLTP and TPO, and therefore are acceptable for TPO.

(4) Condensate Storage and Transfer (CST)

The following piping systems have operating pressures less than design pressures and temperature increases less than or equal to 2°F due to the power increases anticipated for TPO; however, the piping stresses have a minimal increase and remain acceptable for TPO.

(1) Main Steam (MS) (outside containment) including equalization header, turbine bypass piping, and crossover piping

For the MS system piping outside containment, the turbine stop valve (TSV) closure transient was reviewed against conditions that bound operations under TPO as part of the MS system piping analysis described above. Available stress and support load margins are adequate to accommodate the increase in loading associated with this fluid transient.

(2) Bleed Steam (BS) (except as noted below)

(3) Seal Steam (SS)

(5) Heater/ MSR Vents and Drains (HD/HV) (except as noted below)

(6) Reactor FW (RFW) (outside containment)

(7) Condensate (COND)

The following piping systems have temperature increases of greater than 2°F (<2.2°F max) due to the power increases anticipated for TPO; however, the piping stresses have a minimal increase and remain acceptable for TPO.

(2) Bleed Steam (BS) – From the high pressure (HP) turbine to 1st stage reheater and to seal steam evaporators

(5) Heater/ MSR Vents and Drains (HD/HV) – 1st stage reheater drains to 6th stage FW heater; 1st stage reheater vents to 5th stage FW heater

All piping systems analyzed have temperature increases equal or less than 10% of available margin between the design and operating temperature; the piping stresses have a minimal increase and remain acceptable for TPO.

Pipe Supports

For those piping systems that have no change in operating conditions between CLTP and TPO, all the pipe support loads remain unchanged.

For those piping systems that have operating temperatures less than 150°F, temperature increases of less than or equal to 2°F, or temperature increases less than or equal to 10% of available margin due to the power increases anticipated for TPO, pipe support loads will experience a small increase in the thermal load. 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 minimal and remains acceptable.

Therefore, all supports, branch piping and equipment are acceptable for TPO.

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. CGS has an established program for monitoring pipe wall thinning in single phase and two-phase high-energy carbon steel piping. FAC rates may be influenced by changes in fluid velocity, temperature, and moisture content. The FAC monitoring program includes the use of a predictive method to calculate wall thinning of components susceptible to FAC. For TPO, the evaluation of predicted wall thinning of the BOP piping indicates minimal effect.

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 and inspection for FAC in BOP systems is addressed by compliance with Generic Letter (GL) 89-08 (Reference 24). The plant FAC program currently monitors the affected systems. Continued monitoring of the systems provides confidence in the integrity of susceptible high-energy piping systems. Appropriate changes to piping inspection frequency will be implemented to ensure adequate margin exists for those systems with changing process conditions. This action takes into consideration adjustments to predicted material loss rates used to project the need for maintenance/replacement prior to reaching minimum wall thickness requirements. This program provides assurance 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

The reactor recirculation system (RRS) evaluation process is described in TLTR Section 5.6.2. The TPO uprate has a minor effect on the RRS and its components. The TPO uprate does not require an increase in the maximum core flow. No significant reduction of the maximum flow capability occurs due to the TPO uprate because of the small increase in core pressure drop (< 1 psi). 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.

An evaluation has confirmed that no significant increase in RRS vibration occurs due to TPO operating conditions.

3.7 MAIN STEAM LINE FLOW RESTRICTORS

The generic evaluation provided in TLTR Appendix J.2.3.7 is applicable to CGS. The requirements for the MSL flow restrictors remain unchanged for TPO uprate conditions. No change in steam line break flow rate occurs because the operating pressure is unchanged. All safety and operational aspects of the MSL flow restrictors are within previous evaluations.

3.8 Main Steam Isolation Valves

The generic evaluation provided in TLTR Appendix J.2.3.7 is applicable to CGS. The requirements for the main steam isolation valves (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 reactor core isolation cooling (RCIC) system provides inventory makeup to the reactor vessel when the vessel is isolated from the normal high pressure makeup systems. The generic evaluation provided in TLTR Section 5.6.7 is applicable to CGS. 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. The generic evaluation provided in TLTR Section 5.6.4 and Appendices J.2.3.1 and J.2.3.13 are applicable to CGS.

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	Core cooling	See Section 4.2.4
Suppression Pool Cooling (SPC) and Containment Spray Cooling (CSC) Modes	Normal 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 design limit. The CSC mode sprays water into the containment to reduce post-accident containment pressure and temperature.	Containment analyses have been performed at 102% of CLTP.
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	CGS does not have a steam condensing mode of RHR.

Operating Mode	Key Function	TPO Evaluation
Fuel Pool Cooling (FPC) Assist	Supplemental 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, 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

The generic evaluation of the RWCU system provided in TLTR Sections 5.6.6 and J.2.3.4 is applicable to CGS. The performance requirements of the RWCU system are negligibly affected by TPO uprate. There is no significant effect on operating temperature and pressure conditions in the high pressure portion of the system. RWCU flow is not changed for TPO conditions. Steady power level changes for much larger power uprates have shown no effect on reactor water chemistry and the performance of the RWCU system. Power transients that result in crud bursts causing high intermediate loading on the system capacity are the primary source of challenge to the system, so safety and operational aspects of water chemistry performance are not affected by the TPO.

Table 3-1 CGS Upper Shelf Energy 60-Year License (51.56 EFPY)

Material	Heat or Heat/Lot	Internal Traverse USE (ft-lb)	%Cu	51.56 EFPY ¼T Fluence (n/cm^2).	% Decrease USE (1)(2)	51.56 EFPY Traverse USE (3) (ft/lb).
Plates						
Lower Intermediate Shell	B5301-1	98	0.13	7.93E+17	12.5 (4)	86
Welds:						
Vertical:						
Lower Shell						
	3P4966 / 1214-3482 (S) (5)	98	[[ ]] (6)	2.64E+17	7.5	91
	3P4966 / 1214-3482 (T) (5)	98	[[ ]] (6)	2.64E+17	7.5	91
Lower Intermediate Shell						
	3P4966 / 1214-3481 (S) (5)	98	[[ ]] (6)	7.93E+17	9.5	89
	3P4966 / 1214-3481 (T) (5)	98	[[ ]] (6)	7.93E+17	9.5	89
Girth:						
	5P6756 / 0342-3447 (S) (5)	91	[[ ]] (6)	3.22E+17	10	82
	5P6756 / 0342-3447 (T) (5)	97	[[ ]] (6)	3.22E+17	10	87
	5P4955 / 0342-3443 (S) (5)	90	[[ ]] (6)	3.22E+17	8	83
	5P4955/ 0342-3443 (T) (5)	95	[[ ]] (6)	3.22E+17	8	87
Nozzles:						
N6						
Nozzle	Q2Q55W / 790S	70	0.11	4.36E+17	10	63
N6 Weld						
	5P6214B / 0331 (S) (5)	70	[[ ]] (6)	4.36E+17	8 (7)	64
	5P6214B / 0331 (T) (5)	70	[[ ]] (6)	4.36E+17	8 (7)	64
N12						
Nozzle	219972 / 1	62	[[ ]] (8)	2.30E+17	16	52
	718259 / 65363	62	0.25 (8)	2.30E+17	14.5	53
N12 Weld	Inco 82 / 182	NA	NA	NA	NA	NA

Table 3-1 CGS Upper Shelf Energy 60-Year License (51.56 EFPY) continued

Material	Heat or Heat/Lot		Internal Traverse USE (ft-lb)		%Cu		51.56 EFPY ¼ T Fluence (n/cm^2).	% Decrease USE (1)(2)	51.56 EFPY Traverse USE (3) (ft/lb).
ISP Representative Materials (EPRI Proprietary Information (9))									
Plate	B0673-1	(9)(10)	NA		NA		NA	NA	NA
Weld									
River Bend 183 & SSP (F, H, C)	5P6756	(9)(11)	104.4	(9)	0.06	(9)	3.22E+17	9	95
PY1 3 & 177	5P6214B	(9)(11)	90.9	(9)	0.027	(9)	4.36E+17	8.5	83
SSP (A, B, D, E, G, I)	5P6214B	(9)(11)	91.5	(9)	0.01	(9)	4.36E+17	7.5	85

Notes:

1. USE Decrease obtained from Figure 2 in RG 1.99, Revision 2.
2. Rounded up to the nearest 0.5 value.
3. 51.56 EFPY Transverse USE = Initial Transverse USE * [1 - (% Decrease USE /100)].
4. Previous evaluation appears to have used "weld" line vs. "base" line in RG 1.99 Figure 2 calculation, thus resulting in a different % Decrease USE.
5. (S) = Single, (T) = Tandem.
6. Best Estimate Chemistry used; Heat # and %Cu obtained from BWRVIP-135 R3 (Reference 14) (EPRI Proprietary Information).
7. Previous evaluation used conservative %Cu value of 0.05 in RG 1.99 Figure 2 (% decrease USE) calculation. Thus the resulting % decrease USE was higher.
8. Previous evaluation considered different %Cu value.
9. ISP Representative Material Information; Heat #, Initial USE and %Cu obtained from BWRVIP-135 R3 (EPRI Proprietary Information).
10. Representative heat # is not a match to plant material.
11. Representative heat # is a match to plant material.

Table 3-2 RPV Beltline Plate USE Equivalent Margin Analysis (51.56 EFPY)

Equivalent Margin Analysis
Plant Applicability Verification Form for Columbia
Including Power Uprate Conditions
60-Year License
(Cumulative Energy Provided in Fluence Report)
BWR/3-6 Plate

Surveillance Plate USE (Heat B5301-1)

%Cu	=	0.11	
Unirradiated USE	=	98.0 ft-lb	
1st Capsule Measured USE	=	99.6 ft-lb	
1st Capsule Fluence	=	1.55E+17 n/cm^2	
1st Capsule Measured % Decrease	=	-1.6	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	8.0	(RG 1.99, Rev. 2, Figure 2)

ISP Surveillance Plate USE (Heat B0673-1) **(For information purposes only not as a matching beltline heat)**

%Cu	=	0.15	
Unirradiated USE	=	158.1 ft-lb	
DA 288° Capsule Measured USE	=	158.8 ft-lb	
DA 288° Capsule Fluence	=	5.09E+17 n/cm^2	
DA 36° Capsule Measured USE	=	137 ft-lb	
DA 36° Capsule Fluence	=	1.17E+18 n/cm^2	
SSP Capsule F Measured USE	=	133 ft-lb	
SSP Capsule F Fluence	=	1.87E+18 n/cm^2	
DA 108° Capsule Measured USE	=	131.3 ft-lb	
DA 108° Capsule Fluence	=	2.63E+18 n/cm^2	
DA 288° Capsule Measured % Decrease	=	-0.4	(Charpy Curves)
DA 288° Capsule RG 1.99 Predicted % Decrease	=	12.0	(RG 1.99, Rev. 2, Figure 2)
DA 36° Capsule Measured % Decrease	=	13.3	(Charpy Curves)
DA 36° Capsule RG 1.99 Predicted % Decrease	=	14.5	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule F Measured % Decrease	=	15.9	(Charpy Curves)
SSP Capsule F RG 1.99 Predicted % Decrease	=	16.5	(RG 1.99, Rev. 2, Figure 2)
DA 108° Capsule C Measured % Decrease	=	17.0	(Charpy Curves)
DA 108° Capsule C RG 1.99 Predicted % Decrease	=	17.5	(RG 1.99, Rev. 2, Figure 2)

Limiting Beltline Plate USE (Heat C1337-1 and C1337-2):

CLTP (54 EFPY)

%Cu	=	0.15	
54 EFPY 1/4 T Fluence (Cumulative Energy Provided in Fluence Report)	=	8.14E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	13.5	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	NA	(RG 1.99, Rev. 2, Position 2. 2)

TPO (51.56 EFPY)

%Cu	=	0.15	
51.56 with TPO EFPY 1/4 T Fluence (Cumulative Energy Provided in Fluence Report)	=	7.93E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	13.5	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	N/A	(RG 1.99, Rev. 2, Position 2. 2)

13.5% < [[ ]] (for 54 EFPY)

Therefore, vessel plates are bounded by equivalent margin analysis.

Table 3-3 RPV Beltline Plate USE Equivalent Margin Analysis (51.56 EFPY)

Equivalent Margin Analysis
Plant Applicability Verification Form
for Columbia
including Power Uprate Conditions
60-Year License
(Cumulative Energy Provided in Fluence Report)
BWR/2-6 WELD

Surveillance Weld USE (Heat 3P4966):

%Cu	=	0.03	
Unirradiated USE	=	98.0 ft-lb	
1st Capsule Measured USE	=	108.0 ft-lb	
1st Capsule Fluence	=	1.55E+17 n/cm2	
1st Capsule Measured % Decrease	=	-10.2	(Charpy Curves)
1st Capsule RG 1.99 Predicted % Decrease	=	6.0	(RG 1.99, Rev. 2, Figure 2)

ISP Surveillance Weld USE (Heat 5P6756): Potentially use for adjustment because it is a matching be t ine heat.

%Cu	=	0.06	
Unirradiated USE	=	104.4 ft-lb	
River Bend 183° Capsule Measured USE	=	84.4 ft-lb	
River Bend 183° Capsule Fluence	=	1.16E+18 n/cm2	
SSP Capsule F Measured USE	=	79.3 ft-lb	
SSP Capsule F Fluence	=	1.94E+18 n/cm2	
SSP Capsule H Measured USE	=	84.6 ft-lb	
SSP Capsule H Fluence	=	1.58E+18 n/cm2	
SSP Capsule C Measured USE	=	110.7 ft-lb	
SSP Capsule C Fluence	=	2.93E+17 n/cm2	
River Bend 183° Capsule Measured % Decrease	=	19.2	(Charpy Curves)
River Bend 183° Capsule RG 1.99 Predicted % Decrease	=	12.5	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule F Measured % Decrease	=	24.0	(Charpy Curves)
SSP Capsule F RG 1.99 Predicted % Decrease	=	14.0	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule H Measured % Decrease	=	19.0	(Charpy Curves)
SSP Capsule H RG 1.99 Predicted % Decrease	=	13.0	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule C Measured % Decrease	=	-6.0	(Charpy Curves)
SSP Capsule C RG 1.99 Predicted % Decrease	=	9.0	(RG 1.99, Rev. 2, Figure 2)

ISP Surveillance Weld USE (Heat 5P6214B): Potentially use for adjustment because it is a matching be t ine heat.

%Cu	=	0.027
Unirradiated USE	=	90.9 ft-lb
PY1 3° Capsule Measured USE	=	85.8 ft-lb
PY1 3° Capsule Fluence	=	3.18E+17 n/cm2
PY1 177° Capsule Measured USE	=	94.9 ft-lb
PY1 177° Capsule Fluence	=	1.08E+18 n/cm2
%Cu	=	0.01
Unirradiated USE	=	91.5 ft-lb
SSP Capsule D Measured USE	=	89 ft-lb
SSP Capsule D Fluence	=	1.03E+18 n/cm2
SSP Capsule E Measured USE	=	88.5 ft-lb
SSP Capsule E Fluence	=	1.77E+18 n/cm2
SSP Capsule G Measured USE		85.3 ft-lb
SSP Capsule G Fluence		1.95E+18 n/cm2
SSP Capsule I Measured USE		87.7 ft-lb
SSP Capsule I Fluence		2.75E+18 n/cm2
SSP Capsule A Measured USE		96.5 ft-lb
SSP Capsule A Fluence		4.09E+17 n/cm2
SSP Capsule B Measured USE		97.4 ft-lb
SSP Capsule B Fluence		5.26E+17 n/cm2

Table 3-3 (continued)

PY1 3° Capsule Measured % Decrease	=	5.6	(Charpy Curves)
PY1 3° Capsule RG 1.99 Predicted % Decrease	=	7.4	(RG 1.99, Rev. 2, Figure 2)
PY1 177° Capsule Measured % Decrease	=	-4.4	(Charpy Curves)
PY1 177° Capsule RG 1.99 Predicted % Decrease	=	9.9	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule D Measured % Decrease	=	2.7	(Charpy Curves)
SSP Capsule D RG 1.99 Predicted % Decrease	=	8.8	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule E Measured % Decrease	=	3.3	(Charpy Curves)
SSP Capsule E RG 1.99 Predicted % Decrease	=	10.0	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule G Measured % Decrease	=	6.8	(Charpy Curves)
SSP Capsule G RG 1.99 Predicted % Decrease	=	10.2	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule I Measured % Decrease	=	4.2	(Charpy Curves)
SSP Capsule I RG 1.99 Predicted % Decrease	=	11.0	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule A Measured % Decrease	=	-5.5	(Charpy Curves)
SSP° Capsule A RG 1.99 Predicted % Decrease	=	7.0	(RG 1.99, Rev. 2, Figure 2)
SSP Capsule B Measured % Decrease	=	-6.4	(Charpy Curves)
SSP Capsule B RG 1.99 Predicted % Decrease	=	7.5	(RG 1.99, Rev. 2, Figure 2)

Limiting Beltline Weld USE (Heat 624039 / D205A27A):

TPO (51.56 EFPY)

%Cu	=	0.10	
51.56 with TPO EFPY 1/4T Fluence (Cumulative Energy Provided in Fluence Report)	=	7.93E+17 n/cm^2	
RG 1.99 Predicted % Decrease	=	13.5	(RG 1.99, Rev. 2, Figure 2)
Adjusted % Decrease	=	20.0	(RG 1.99, Rev. 2, Position 2. 2)

20.0% < [[ ]] (for 54 EFPY)

Therefore, vessel welds are bounded by equivalent margin analysis

Table 3-4 CGS Adjusted Reference Temperatures 60-Year License (51.56 EFPY)

Component	Heat or Heat Lot (1)	%Cu	%Ni	CF (2)	Adjusted CF (3)	Initial RTndt °F	¼T Fluence n/cm²	51.56 EFPY ΔRTndt °F	σ_I	σ_Δ	Margin °F	51.56 EFPY Shift °F	51.56 EFPY ART °F
PLATES:													
Lower Shell													
Mk 21-1-1	C1272-1	0 15	0 60	110		28	2 64E+17	22 4	0	11 2	22 4	44 8	72 8
Mk 21-1-2	C1273-1	0 14	0 60	100		20	2 64E+17	20 4	0	10 2	20 4	40 7	60 7
Mk 21-1-3	C1273-2	0 14	0 60	100		4	2 64E+17	20 4	0	10 2	20 4	40 7	44 7
Mk 21-1-4	C1272-2	0 15	0 60	110		0	2 64E+17	22 4	0	11 2	22 4	44 8	44 8
Shell # 2													
Mk 22-1-1	B5301-1	0 13	0 50 (4)	88		-20	7 93E+17	32 8	0	16 4	32 8	65 5	45 5
Mk 22-1-2	C1336-1	0 13	0 50	88		-8	7 93E+17	32 8	0	16 4	32 8	65 5	57 5
Mk 22-1-3	C1337-1	0 15	0 51	105		-20	7 93E+17	39 1	0	17 0	34 0	73 1	53 1
Mk 22-1-4	C1337-2	0 15	0 51	105		-20	7 93E+17	39 1	0	17 0	34 0	73 1	53 1
NOZZLES:													
N6													
Nozzle	Q2Q55W / 790S	0 11	0 76	76		-20	4 36E+17	20 8	0	10 4	20 8	41 5	21 5
Weld	5P6214B/0331 (S) (5, 6, 7)	[[ ]] (6)	[[ ]] (6)	27	[[ ]] (5, 8)	-50 (9, 10)	4 36E+17	15 9	0	7 9	15 9	31 7	-18 3
Weld	5P6214B/0331 (T) (5, 6, 7)	[[ ]] (6)	[[ ]] (6)	27	[[ ]] (5, 8)	-24 (9, 10)	4 36E+17	15 9	0	7 9	15 9	31 7	7 7
N12													
Nozzle	219972/1 (11)	[[ ]] (12, 13)	[[ ]] (12, 13	[[ ]]		40	2 30E+17	25 5	0	12 8	25 5	51 1	91 1
Nozzle	718259/65363 (11)	0 25 (16)	0 24 (16)	131		-20	2 30E+17	24 5	0	12 3	24 5	49 1	29 1
Weld	Inco 82/182	(17)											
WELDS:													
Vertical													
Lower Shell													
BA, BB, BD	04P046 / D217A27A	0 06	0 90	82		-48	2 64E+17	16 7	0	8 4	16 7	33 4	-14 6
BA, BB	07L669 / K004A27A	0 03	1 02	41		-50	2 64E+17	8 4	0	4 2	8 4	16 7	-33 3
BA, BB, BC, BD	3P4966 / 1214 – 3482 (S) (6)	[[ ]] (6)	[[ ]] (6)	34 (6)		-30	2 64E+17	6 9	0	3 5	6 9	13 9	-16 1
BA, BB, BC, BD	3P4966 / 1214 – 3482 (T) (6)	[[ ]] (6)	[[ ]] (6)	34 (6)		-48	2 64E+17	6 9	0	3 5	6 9	13 9	-34 1
BB, BC, BD	C3L46C / J020A27A	0 02	0 87	27		-20	2 64E+17	5 5	0	2 8	5 5	11 0	-9 0
BB	08M365 / G128A27A	0 02	1 10	27		-48	2 64E+17	5 5	0	2 8	5 5	11 0	-37 0
BC	09L853 / A111A27A	0 03	0 86	41		-50	2 64E+17	8 4	0	4 2	8 4	16 7	-33 3
Lower-Intermediate Shell													
BE,BF, BG, BH													
BE,BF, BG, BH	3P4966 / 1214 – 3481 (S) (6)	[[ ]] (6)	[[ ]] (6)	34 (6)		-20	7 93E+17	12 7	0	6 3	12 7	25 3	5 3
BF, BH	3P4966 / 1214 – 3481 (T) (6)	[[ ]] (6)	[[ ]] (6)	34 (6)		-6	7 93E+17	12 7	0	6 3	12 7	25 3	19 3
BF	04P046 / D217A27A	0 06	0 90	82		-48	7 93E+17	30 5	0	15 3	30 5	61 0	13 0
BG	05P018 / D211A27A	0 09	0 90	122		-38	7 93E+17	45 4	0	22 7	45 4	90 8	52 8
BG	624063 / C228A27A	0 03	1 00	41		-50	7 93E+17	15 3	0	7 6	15 3	30 5	-19 5
BH	624039 / D224A27A	0 07	1 01	95		-36	7 93E+17	35 4	0	17 7	35 4	70 7	34 7
	624039 / D205A27A	0 10	0 92	134		-50	7 93E+17	49 9	0	24 9	49 9	99 7	49 7
Girth													
AB													
AB	492L4871 / A422B27AF	0.03	0 98	41		-50	3 22E+17	9 4	0	4 7	9 4	18 8	-31 2
AB	04T931 / A423B27AG	0 03	1 00	41		-50	3 22E+17	9 4	0	4 7	9 4	18 8	-31 2
AB	5P6756 / 0342-3447 (S) 6, 7)	[[ ]] (6)	[[ ]] (6)	108 (6)	[[ ]] (14)	-50	3 22E+17	35 2	0	14 0 (15)	28 0	63 2	13 2
AB	5P6756 / 0342-3447 (T) (6, 7	[[ ]] (6)	[[ ]] (6)	108 (6)	[[ ]] (14)	-50	3 22E+17	35 2	0	14 0 (15	28 0	63 2	13 2
AB	3P4955 / 0342-3443 (S) (6)	[[ ]] (6)	[[ ]] (6)	37 (6)		-16	3 22E+17	8 5	0	4 2	8 5	16 9	0 9
	3P4955 / 0342-3443 (T) (6)	[[ ]] (6)	[[ ]] (6)	37 (6)		-20	3 22E+17	8 5	0	4 2	8 5	16 9	-3 1

Table 3-4 (continued)

Notes:

1. For weld materials, S = Single Wire, T = Tandem Wire.
2. Determined per RG 1.99 Tables 1 and 2.
3. Adjusted CF obtained per Notes 5, 8 & 14.
4. B5301-1 is the surveillance plate material; this chemistry represents the average of the CGS Surveillance Capsule Report (Reference 21) chemistry test results averaged with the baseline CMTR values.
5. Adjusted CF is based on best estimate chemistry data.
6. Best estimate chemistry data from BWRVIP-135 R3 considered.
7. Chemistry adjusted to consider ISP date from BWRVIP-135 R3.
8. [[ ]]
9. Initial RT_{NDT} calculated from plant-specific CMTRs, using the GEH method.
10. The previous PT curve evaluation did not have sufficient information to determine plant specific RT_{NDT}, so it conservatively calculated Initial RT_{NDT} from BWRVIP-135 R3 (Page 3-14) for the case where the Initial RTNDT is not available.
11. Material = 508-CL 1.
12. Cu and Ni values listed are from GEH internal 508-CL 1 surveillance study 0000-0151-4184-R0.
13. The previous PT curve evaluation did not have sufficient information to determine plant specific chemistry; therefore, maximum %Cu and %Ni values were previously conservatively considered.
14. [[ ]]
15. Credible data so 1/2 margin term Per BWRVIP-135 for sigma(Δ).
16. CU and Ni values listed are from CMTR.
17. Inco 82/182 does not require Fracture toughness evaluation.

Table 3-5 CGS 51.56 EFPY Effects of Irradiation on RPV Axial Weld Properties

Axial Weld Parameters at 51.56 EFPY

Parameter	NRC Limiting Plant Specific Analysis (Axial Welds) [1] 64 EFPY (CB&I RPV)	CGS ISP Chemistry Limiting Weld Wire (05P018) 51.56 EFPY (CB&I RPV)
Cu%	0.10	0.09
Ni%	1.08	0.90
CF	135	122
End of License Inside Diameter Fluence, (10^{19} n/cm^2)	1.38	0.115
$RT_{NDT(U)}$ (°F)	-30	-38
ΔRT_{NDT} w/o margin (°F) [2]	147.1	54.3
Mean RT_{NDT} (°F)	117.1	16.3
P (F/E) [3] NRC	3.82E-01	[5]
P (F/E) [6] SER BWRVIP-74	5.00E-06	[6]

Notes:

1. Chemistry information reported in BWRVIP-05
2. $\Delta RT_{NDT} = CF * f^{(0\ 28 - 0\ 10 \log f)}$
3. P (F/E) stands for "Probability of a failure event".
4. The mean RT_{NDT} value from Table 1 of the SER for BWRVIP-74 that corresponds to a failure frequency of 5.0E-6 (For Pilgrim, a BWR3) is 114°F.
5. Although a conditional failure probability has not been calculated, the fact that the CGS values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the CGS RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL98-05.
6. The CGS axial weld mean RT_{NDT} remains well below the 114 °F from the SER for BWRVIP-74, thus CGS axial weld failure frequency is well below the acceptable limit of 5.0E-06.

Table 3-6 CGS 51.56 EFPY Effects of Irradiation on RPV Circumferential Weld Properties

Circumferential Weld Parameters at 51.56 EFPY

Parameter	NRC Limiting Plant Specific Analysis (Circ Welds) (1) 64 EFPY (CB&I RPV)	CGS Overall Limiting Weld Wire (3P4955) 51.56 EFPY (CB&I RPV)
Cu%	0.10	[[ ]]
Ni%	0.99	[[ ]]
CF	134.9	37
End of license inside diameter fluence (10^{19} n/cm^2)	1.02	0.047
$RT_{NDT(U)}$ (°F)	-65	-16
ΔRT_{NDT} without margin (°F) (2)	135.6	10.4
Mean RT_{NDT} (°F)	70.6	-5.6
P (F/E) NRC (3)	1.78E-05	(4)

Notes:

1. Chemistry information reported in BWRVIP-05.
2. $\Delta RT_{NDT} = CF * f^{(0 28 - 0 10 \log f)}$
3. P (F/E) stands for "Conditional probability of vessel failure or probability of vessel failure assuming that the event occurred".
4. Although a conditional failure probability has not been calculated, the fact that the CGS values at the end of license are less than the 64 EFPY value provided by the NRC leads to the conclusion that the CGS RPV conditional failure probability is bounded by the NRC analysis, consistent with the requirements defined in GL98-05.

Table 3-7 CUF and P+Q Stress Range of Limiting Components

	P + Q Stress (ksi)			CUF(4,5)		
Component(1,6)	Current (3,486 MWt)	TPO (3,545 MWt)(3)	Allowable (ASME Code Limit)	Current (3,486 MWt)	TPO (3,545 MWt)(3)	Allowable (ASME Code Limit)
Shroud Support	70.74/ 65.58(2)	70.74/ 65.58(2,6)	69.9	0. 399	0. 399	1.0
Steam Dryer Bracket	38.4	38.4	42.3	0.064	0.064	1.0

Notes:

1. There are no changes in operating conditions from CLTP to TPO. Therefore, the CLTP evaluation remains applicable for TPO. The components presented in this table are consistent with the CLTP SAR to demonstrate that the results remain unchanged from CLTP to TPO.
2. Thermal bending included/thermal bending removed. P + Q stresses are acceptable per CLTP elastic-plastic analysis where applicable, which is valid for TPO conditions.
3. [[

]]
4. Limiting CUF is presented.
5. Fatigue usage factors are for a 40-year license.
6. CLTP and TPO were [[ ]] Therefore, there is no change in values from CLTP to TPO.

Table 3-8 Governing Stress Results for RPV Internals

Item	Component	Location	Service Level	Stress Category	Unit	CLTP Value	TPO Value(1)	Allowable Limit(2)
1	Shroud Support(7)	Legs	B	P_m+P_b	psi	25,540	25,540	28,100
2	Shroud(7)	Top Guide Wedge	B	P_m+P_b	psi	12,730	12,730	21,450
3	Core Plate	Longest Beam	B	Buckling	lbs/ CRGT	1,016	1,016	1,179
4	Top Guide	Longest Beam	B	P_m+P_b	psi	28,548	28,548	31,690
5a	Control Rod Drive Housing (Outside RPV Portion)	CRD Housing at RPV Bottom Head	B	P_m+P_b	psi	15,450	15,450	24,900
5b	Control Rod Drive Housing (Inside RPV Portion)	CRD Housing at RPV Stub Tube	B	P_m+P_b	psi	11,925	11,925	16,185
5c	Control Rod Drive Mechanism	CRD Outer Tube	B	P_m+P_b	psi	24,700	24,700	26,100
6a	Control Rod Guide Tube	CRGT Flange (Base)	B	P_m+P_b	psi	8,189	8,189	24,000
6b	Control Rod Guide Tube	Mid-Span	B	P_m+P_b	psi	9,100	9,100	16,000
6c	Control Rod Guide Tube	Body	B	Buckling	N/A	0.40	0.40	0.45
7	Orificed Fuel Support	OFS Body	B	Load	lbs	14,895(3)	14,895(3)	35,590(3)

Table 3-8 Governing Stress Results for RPV Internals (continued)

Item	Component	Location	Service Level	Stress Category	Unit	CLTP Value	TPO Value(1)	Allowable Limit(2)
8	Feedwater Sparger	FW Pipe-to-Sparger Weld	B	Fatigue Usage	N/A	0.88(4)	0.48(5)	1.0
9	Jet Pump Assembly(7)	Riser Brace	D	P_m+P_b	psi	54,427	54,427	60,840
10a	Core Spray Line	Elbow	B	P_m+P_b	psi	19,890	19,890	23,850
10b	Core Spray Sparger	Tee Junction	B	P_m	psi	6,560	6,560	21,450
11	Access Hole Cover(7) (Top Hat Design)	Cover	D	P_m+P_b	psi	16,031	37,352	49,400
12	Shroud Head and Steam Separator Assembly	Shroud Head Bolt	B	P_m	psi	7,909	7,909	16,900
13	In-Core Housing and Guide Tube	In-Core Housing at RPV Penetration	B	P_m+P_b	psi	25,160	25,160	25,400
14	Core Differential Pressure and Liquid Control Line	Unknown	C	P_m+P_b	psi	17,015	17,015(6)	36,900
15	Low Pressure Coolant Injection Coupling	Support Ring	C	P_m+P_b	psi	27,600	27,600	31,400

Notes:

(1) Stresses/loads listed are for the limiting loading condition, with the least margin of safety.
(2) Allowable values are consistent with the original design basis.
(3) For OFS, the calculated and allowable loads are in the vertical downward direction.
(4) Based on a generic GEH FW sparger analysis report. Includes conservative assumptions.
(5) Based on a 60-year plant life and also based on a generic GEH FW sparger analysis report. However, the conservatism was removed from the generic FW sparger analysis.
(6) For the core differential and liquid control line, the calculated stress shown is based on an absolute summation of upset loads. Actual stress based on the square root sum of squares (SRSS) methodology will be less.
(7) These components are affected by GEH SCs.

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 current 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.

The following table summarizes the effect of the TPO uprate on various aspects of the containment system performance.

<table>
<tr><th>Topic</th><th>Key Parameters</th><th>TPO Effect</th></tr>
<tr><td>Short Term Pressure and Temperature Response</td><td></td><td rowspan="11">Current Analysis
Based on 102% of CLTP</td></tr>
<tr><td>Gas Temperature</td><td>Break Flow and Energy</td></tr>
<tr><td>Pressure</td><td>Break Flow and Energy</td></tr>
<tr><td>Long-Term Suppression Pool Temperature Response</td><td></td></tr>
<tr><td>Bulk Pool</td><td>Decay Heat</td></tr>
<tr><td>Local Temperature with SRV Discharge</td><td>Decay Heat</td></tr>
<tr><td>Containment Dynamic Loads</td><td></td></tr>
<tr><td>Loss-of-Coolant Accident Loads</td><td>Break Flow and Energy</td></tr>
<tr><td>Safety-Relief Valve Loads</td><td>Decay Heat</td></tr>
<tr><td>Sub-compartment Pressurization</td><td>Break Flow and Energy (Note 1)</td></tr>
<tr style="display:none"><td></td><td></td></tr>
<tr><td>Containment Isolation
Section 4.1.1 provides confirmation that motor-operated valves (MOVs) are capable of performing design basis functions at TPO conditions.</td><td></td><td>The ability of containment isolation valves (CIVs) and operators to perform their required functions is not affected because the evaluations have been performed at 102% of CLTP.</td></tr>
</table>

Note:

1. The CGS current analysis of sub-compartment pressurization is based on the maximum break flow and energy of postulated pipe breaks between the RPV wall and the biological shield wall. GEH recently issued safety information communication SC 09-01 (Reference 25) to

inform BWR plant owners of possible non-conservative assumptions in the current analysis. Sub-compartment pressurization is not affected by the TPO operating conditions because the current analysis operating conditions with a two-percent power measurement uncertainty bounds the TPO with the improved power measurement uncertainty. The issues identified in SC 09-01 have been reviewed and resolved for CGS.

4.1.1 Generic Letter 89-10 Program

The motor-operated valve (MOV) requirements in the FSAR were reviewed, and no changes to the functional requirements of the GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," MOVs, were identified as a result of operating at the TPO RTP level. Because previous analyses were either based on 102% of CLTP or are consistent with the plant conditions expected to result from TPO, there are no increases in the pressure or temperature at which MOVs are required to operate with the exception of RFW valves (slight temperature increase, but no field modifications required). 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," was reviewed and determined to have no effects related to this power 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," 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 CGS response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions," was reviewed for the TPO uprate. The containment design temperatures and pressures in the current GL 96-06 evaluation are not exceeded under post-accident conditions for the TPO uprate. Therefore, the CGS response to GL 96-06 remains valid under TPO uprate conditions.

4.1.5 Containment Coatings

The nominal operating conditions change slightly and the required initial conditions for containment analysis inputs remain the same for TPO. The temperature and pressure do not increase significantly. The Service Level 1 coatings are qualified to 340°F, 70 psi and 1.1×10^6 rads. 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 high pressure coolant injection (HPCI) system is not applicable to CGS.

4.2.2 High Pressure Core Spray

The high pressure core spray (HPCS) system is a motor 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 HPCS 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 generic evaluation of the HPCS system provided in TLTR Section 5.6.7 is applicable to CGS. The ability of the HPCS system to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the HPCS system are within previous evaluations and the requirements are unchanged for TPO uprate conditions.

4.2.3 Low Pressure Core Spray

The low pressure core spray (LPCS) system sprays water into the reactor vessel after it is depressurized. The primary purpose of the LPCS 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. The generic evaluation of the LPCS system provided in TLTR Section 5.6.10 is applicable to CGS. The ability of the LPCS system to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the LPCS system are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

4.2.4 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. The generic evaluation of the LPCI mode provided in TLTR Section 5.6.4 is applicable to CGS. The ability of the RHR system to perform required safety functions required by the LPCI mode is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the RHR system LPCI mode are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

4.2.5 Automatic Depressurization System

The 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 LPCS and LPCI to inject coolant into the RPV. The ADS initiation logic and valve control is not affected by the TPO uprate. The generic evaluation of the ADS provided in TLTR Section 5.6.8 is applicable to CGS. The ability of the ADS to perform required safety functions is demonstrated with previous analyses based on 102% of CLTP. Therefore, all safety aspects of the ADS are within previous evaluations and the requirements are unchanged for the TPO uprate conditions.

4.2.6 ECCS Net Positive Suction Head

The most limiting case for NPSH typically occurs at the peak long-term suppression pool temperature. The generic evaluation of the containment provided in TLTR Appendix G is applicable to CGS. The CLTP containment analyses were based on 102% of CLTP and there is no change in the available NPSH for systems using suppression pool water. Therefore, the TPO uprate does not affect compliance with the ECCS pump NPSH requirements.

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 CGS have been performed at power levels up to 106% and therefore bounds 102% of CLTP, consistent with Appendix K. Table 4-1 shows the results of the CGS ECCS-LOCA analysis. The ECCS-LOCA results for CGS are in conformance with the licensing requirements of 10 CFR 50.46. Therefore, the pre-TPO LOCA analysis for GNF2 fuel bounds the 1.66% TPO uprate for CGS.

Reference 17 provides justification for the GNF2 elimination of the 1,600°F upper bound PCT limit and generic justification that the licensing basis PCT will be conservative with respect to the upper bound PCT. The NRC SER for Reference 17 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 26 provided justification for the elimination of the upper bound PCT limit for CGS.

For the TPO uprate there are no changes to the plant configuration that would invalidate the Reference 26 CGS LOCA evaluation for conformance with Reference 17.

The pre-TPO LOCA analysis for GNF2 fuel is concluded to bound the 1.66% TPO uprate for CGS.

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 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM

CGS does not have a MSIV leakage control system.

4.7 POST-LOCA CONTAINMENT ATMOSPHERE CONTROL SYSTEM

The containment atmosphere control (CAC) system was designed to maintain the post-LOCA concentration of oxygen or hydrogen in the containment atmosphere below the flammability limit. However, since the CGS containment is inerted with nitrogen during plant operation, the CAC system is not required. The CAC system was permanently deactivated by Plant Design Change (PDC) 3539 in 2005.

Table 4-1 CGS ECCS-LOCA Analysis Results for GNF2 Fuel

Parameter	MELLLA	Analysis Limit
Nominal PCT	1,356°F	N/A
Appendix K PCT	1,637°F	≤2,200°F (1)
Licensing Basis PCT	1,700°F	≤2,200°F (1)
Maximum Local Oxidation	<1.0%	≤ 17% (1)
Core-Wide Metal-Water Reaction	<0.1%	≤ 1.0% (1)

Note:

1. 10 CFR 50.46 ECCS-LOCA analysis acceptance criteria

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, Intermediate Range Monitors, and Source Range Monitors

The average power range monitors (APRMs) are re-calibrated to indicate 100% at the TPO RTP level of 3,544 MWt. The APRM high flux scram and the upper limit of the 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. This approach for the CGS TPO uprate follows the guidelines of TLTR Section 5.6.1 and Appendix F, which is consistent with the practice approved for GE BWR uprates in ELTR1 (Reference 2).

For the TPO uprate, no adjustment is needed to ensure the intermediate range monitors (IRMs) have adequate overlap with the source range monitors (SRMs) and APRMs. However, normal plant surveillance procedures may be used to adjust the IRM overlap with the SRMs and the APRMs. The IRM channels have sufficient margin to the upscale scram trip on the highest range when the APRM channels are reading near their downscale alarm trip because the change in APRM scaling is so small 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 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.

5.1.1.3 Rod Block Monitor

The rod block monitor (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. All instrumentation with control functions has sufficient range/adjustment capability for use at the TPO uprate conditions. No safety-related BOP system setpoint changes are required as a result of the TPO uprate. The plant-specific instrumentation and control design and operating conditions are bounded by those used in the evaluations contained in the TLTR.

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. The turbine utilizes a digital electro hydraulic (DEH) control system consisting of solid state governing devices, startup control devices, emergency devices for turbine protection, and special control and test devices. The system operates the high-pressure turbine throttle valves and governor valves, turbine bypass valves, reheater stop and intercept valves, and other protective devices.

Satisfactory reactor pressure control by the turbine pressure regulator and the turbine control valves (TCVs) requires an adequate flow margin between the TPO RTP operating condition and the steam flow capability of the TCVs at their maximum stroke (i.e., valves wide open (VWO)). CGS has demonstrated acceptable pressure control performance at current rated conditions and has in excess of the ~2% steam flow margin needed for the TPO uprate. The existing DEH control system, as designed for the current 100% CLTP conditions, is adequate and require no electronic component changes for the TPO uprate conditions.

No modification is required for the turbine bypass valves. 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 to operate optimally at the TPO uprate power level.

PCS tests, consistent with the guidelines in TLTR Appendix L, will be performed during the power ascension phase.

5.2.2 DEH Turbine Control System

The DEH system was discussed in Section 5.2.1. The existing DEH control system, as designed for the current 100% CLTP conditions, is adequate and requires no electronic component changes for the TPO uprate conditions.

5.2.3 Feedwater Control System

An evaluation of the ability of the FW level control system and FW TCVs, and/or FW turbine controls to maintain adequate water level control at the TPO uprate conditions has been performed. The ~2% increase in FW flow associated with TPO uprate 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. Per the guidelines of TLTR Appendix L, the performance of the FW level control systems will be recorded at 95% and 100% of CLTP and confirmed at the TPO power during power ascension. These checks will demonstrate acceptable operational capability and will utilize the methods and criteria described in the original startup testing of these systems.

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 potentially could be 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. As described in TLTR Section F.4.2.8, 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.

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 18 and 19*)* are used to generate the allowable values (AVs) and nominal trip setpoints (NTSPs) related to any Analytical Limit (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 5-1 lists the ALs (or AVs 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 allowable value (AV) and nominal trip setpoint (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) Sections 5.8 and F.4 and on Section 5.3 of Reference 18.

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

The AL for the turbine hydraulic pressure (low oil pressure trip) that initiates the T/G trip scram 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 core flow 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. For CGS, the AL of 140% of rated steam flow is not changed and no new instrumentation is required (the existing instrumentation has the required upper range limit and calibrated span on the instrument loops to accommodate the new setpoint). A new setpoint was calculated using the GEH methodology per Reference 19 and a TS AV change is required to change the differential pressure at the allowable steam flow.

Because of the large spurious trip margin, sufficient margin to the trip setpoint exists to allow for normal plant testing of the MSIVs. This is consistent with TLTR Section F.4.2.5.

5.3.6 Fixed APRM Scram

The fixed APRM ALs, for both two (recirculation) loop (TLO) and SLO, expressed in percent of RTP do not change for the TPO uprate. The generic evaluation and guidelines presented in TLTR Section F.4.2.2 are applicable to CGS. The limiting transient that relies on the fixed APRM trip is the vessel overpressure transient (main steam isolation valve closure (MSIVC)) with indirect scram. This event has been analyzed assuming 102% of CLTP and is reanalyzed on a cycle specific basis.

5.3.7 APRM Simulated Thermal Power Scram

The simulated thermal power (STP) APRM AVs, 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 RTP, they decrease in proportion to the power uprate or CLTP RTP/TPO RTP. This is the same approach taken for generic BWR uprates described in ELTR1 (Reference 2). There is no significant effect on the instrument errors or uncertainties from the TPO uprate. Therefore, the NTSP is established by directly incorporating the change in the AV.

5.3.8 Rod Worth Minimizer Low Power Setpoint

The rod worth minimizer (RWM) low power setpoint (LPSP) is used to enforce the rod patterns established for the control rod drop accident at low power levels. The TPO RWM LPSP AL of 10% of RTP is not changed. It is conservative to keep the existing percent of rated power after TPO uprate. The generic guidelines in TLTR Section F.4.2.9 are applicable to CGS.

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

CGS does not have a flow-biased RBM system.

5.3.11 Main Steam Line High Radiation Isolation

Deleted per License Amendment 112 (Reference 20).

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. This AL is not changed for the TPO as discussed in TLTR Section F.4.2.7.

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 TS setpoints for scram, high-pressure spray, 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 are not changed for the TPO uprate.

Water level change during operational transients (e.g., trip of a recirculation pump, FW controller failure, loss of one FW pump) 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 above, the high steam tunnel temperature AL 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 (turbine trip / MSIVC).

5.3.16 TSV Closure Scram, TCV Fast Closure Scram Bypasses

The turbine first-stage pressure (TFSP) bypass allows the turbine stop valve (TSV) closure scram and TCV fast closure scram to be bypassed, when reactor power is sufficiently low, such that the scram 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 and TCV fast closure scram bypass AL in percent of RTP is reduced by the ratio of the power increase. The new AL does not change with respect to absolute thermal power. [[
]] The maneuvering range for plant startup is maximized.

No modifications to the CGS turbine are made for the TPO uprate, so there is no change in the first-stage pressure/steam flow relationship from previous operation.

Table 5-1 Analytical Limits and Allowable Values for Current and TPO Power Level

Parameter	Current	TPO	Justification
APRM High Neutron Flux-Fixed Scram (% RTP), AL	123	No change	
APRM STP – High (Scram) (2)			(1)
STP-High Scram Clamp (%RTP) (4), AV	114.9	No change	
TLO STP-High Scram (%RTP) (3), AV	$0.63W_d + 64.0$	$0.62W_d + 62.9$	(4)
SLO STP-High Scram (%RTP) (3), AV	$0.63(W_d - \Delta W) + 64.0$ $0.63W_d + 60.8$	$0.62(W_d - \Delta W) + 62.9$ $0.62W_d + 59.8$	(4)
APRM STP – High (Rod Block) (2)			(1)
STP Rod Block Clamp (%RTP) (4), AV	111	No change	
TLO STP-High Rod Block (%RTP) (3), AV	$0.63W_d + 60.1$	$0.62W_d + 59.1$	(4)
SLO STP-High Rod Block (%RTP) (3), AV	$0.63(W_d - \Delta W) + 60.1$ $0.63W_d + 56.9$	$0.62(W_d - \Delta W) + 59.1$ $0.62W_d + 56.0$	(4)
TSV & TCV Closure Scram Bypass - AL (%RTP)	30	29.5	(5)
MSL High Flow Isolation – ALs: % rated steam flow psid	 140 127.5	 No change 145.37	(5)
Rod Worth Minimizer LPSP – AL (%RTP)	10	No change	(5)

Notes:

(1) CGS does not have ALs for these setpoint functions.

(2) No credit is taken in any safety analysis for flow biased setpoints.

(3) W_d is % recirculation drive flow where 100% drive flow is that required to achieve 100% core flow at 100% power and ΔW is the difference in % drive flow between the TLO and SLO recirculation drive flow at the same core flow.

(4) These changes to the AVs are based upon the methodology approved by the NRC in Reference 1.

(5) All limits scaled for an uprate of 1.66% thermal power.

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 AC POWER

The plant electrical characteristics at TPO uprated conditions 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 uprated 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 generator is a direct-driven 3-phase 60 Hz, 25,000 V, 1,800 rpm, hydrogen inner-cooled, synchronous generator rated for: 1,230 megavolt amps (MVA) at a 0.975 power factor (PF), with a 0.58 short circuit ratio at a nominal hydrogen pressure of 75 psig.
- Main Transformers: The 1,276/1,425 MVA MPT consists of three single-phase,500 - 25 kilo volt (kV), oil directed, air forced (ODAF), 55°C/65°C rise, 60 Hz, oil-filled type, outdoor transformer.
- Isolated Phase Bus Duct: The isolated phase bus duct consists of a main bus, 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 self-cooling for the delta and auxiliary buses. The main bus is rated at 30,000 A with a momentary fault current rating of 215,000 A. The delta bus is rated at 17,300 A with a momentary fault current rating of 215,000 A. The auxiliary bus subsections are rated at 1,200 A with a momentary fault current rating of 385,000 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 1,030,000 Btu/hr.

The review of the existing off-site electrical equipment concluded the following:

- The main generator will be operating within the existing generating capability curve for TPO uprate. For summer and winter operations, the gross generator MWe output is on the existing generator capability curve at close to unity power factor.
- 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 1.66% 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 except if the

spare main transformer is placed into service. There is a potential for a plant downpower if transformer cooling cannot be maintained at 100% plant loading.

A grid stability analysis has been performed, considering the increase in electrical output, to demonstrate conformance to General Design Criteria (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. The grid stability details are provided in a separate enclosure to the License Amendment Request submittal.

6.1.2 On-Site Power

The on-site distribution system loads were reviewed under normal and emergency operating scenarios. The loads are computed based on equipment nameplate ratings. These loads are used as inputs for the computation of anticipated maximum running current, voltage drop, and short circuit currents. Operation at the TPO RTP level is achieved by operating equipment at or below the nameplate rated brake horsepower (BHP). Therefore, there are minimal changes to the calculated equipment loading, system voltage drop or short circuit current values.

The only identifiable changes in electrical load demand are associated with the condensate pumps and condensate booster pumps. These pumps experience increased flow due to the TPO uprate conditions. As a result, each pump motor experiences an increase in load current due to the TPO uprate conditions. The increase in load current resulting from the BHP demands on the condensate pumps, due to the TPO conditions, are still within the equipment nameplate ratings. Therefore, the condensate pump motors and the upstream auxiliary equipment are adequate to support the 68%, 100% and 110% TPO conditions.

The increase in load current resulting from the BHP demands on the condensate booster pump motors, due to the TPO conditions, are not within the equipment nameplate ratings. The condensate booster pump motors experience an overloaded condition of 1.20% above their nameplate ratings due to the TPO conditions. Therefore, the condensate booster pump motors are adequate to support only the 68% and 100% TPO conditions. The 110% transient condition occurs between once per calendar year to once in a lifetime for 59 seconds. The condensate booster pump motor insulation is tested every two years and refurbished every ten years; therefore, the condensate booster pump motors are monitored by the plant sufficiently to prevent concerns of motor operation for the 110% transient condition.

Although the condensate booster pump motors will experience an overload of 1.20% above their nameplate rating, the changes to the on-site alternating current (AC) power system design basis loads, voltage regulation or reduction in design margins due to the TPO conditions are minimal.

The system environmental design basis is unchanged. Operation at the TPO RTP level is achieved by ensuring that sufficient margin exists in the existing equipment operating at the uprated BHP requirements. Under normal conditions, the electrical supply and distribution components (e.g., switchgears and cables) are adequate.

6.2 DC POWER

The 125 VDC non-divisional direct current (DC) electrical distribution system as described in the FSAR, design basis document, electrical loading calculations and drawings was reviewed and no loads that are dependent on reactor power were identified. The non-divisional DC electrical distribution system provides instrumentation, control and motive power for various systems and components. The operation at the TPO RTP level does not increase any load or revise any control logic. Therefore, there are no changes to the load, voltage drop, and short-circuit current values of the 125 VDC non-divisional system.

6.3 FUEL POOL

The following sections address fuel pool cooling (FPC), 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 design limits of the system and its components. The FPC system meets the FSAR requirements at the TPO conditions.

6.3.1 Fuel Pool Cooling

The spent fuel pool (SFP) heat load remains within the capability of the FPC system as assured 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 FPC system supplemented with RHR assist mode, as shown in Table 6-6. The TPO heat load is within the design basis heat load for the FPC system 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 FPC system to maintain adequate FPC for the TPO uprate.

The FPC system 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 FPC assist mode is available to maintain the SFP water temperature below the design limit.

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 FPC system.

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

6.4.1 Service Water Systems

6.4.1.1 Safety-Related Loads

Service Water (SW)

The SW system is the safety-related support system designed primarily to remove reactor decay heat during periods of normal shutdown by providing a heat sink for the RHR system and to provide a heat sink for emergency plant equipment during and after transient and/or accident conditions.

The SW design and ultimate heat sink (UHS) analysis were based on 104.3% of CLTP and bounds the analytical power level for a 1.66% TPO uprate. The increases in the heat loads to equipment cooled by SW are within the existing capacity of the SW system.

6.4.1.2 Non-Safety Related Loads

The power-dependent heat loads on the plant service water (TSW) system that are increased by the TPO are those related to the operation of heat exchangers associated with the generator and auxiliaries equipment, caused by increased T/G output. These include the generator hydrogen coolers, main turbine lube oil coolers, generator stator water coolers, exciter air coolers, seal oil coolers, and isophase bus duct coolers. The condensate pump motor coolers and condensate booster pump lube oil coolers are also affected by increased reactor power due to connections with FW supply to the reactor. The remaining TSW system heat loads are not strongly dependent upon reactor power and do not significantly increase.

The major operational heat load increases to the TSW system from the TPO reflect an operational increase in main generator losses rejected to the generator hydrogen coolers and the main turbine lube oil coolers. The total resulting design heat loads to the TSW system are less than 2% above CLTP. The increases in heat loads to equipment cooled by the TSW system are minimal, and the design capacities of the system and components are adequate to accommodate TPO for power increase. The TSW system has sufficient capacity to assure that adequate heat removal capability is available for TPO operation.

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 the 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 state-issued National Pollutant Discharge Elimination System (NPDES) permit for CGS (WA002515-1, effective November 1, 2014) provides the effluent limitations and monitoring requirements for wastewater discharges to the Columbia River. Specifically, circulating cooling water blowdown quantity is limited to a daily maximum of 9.4 million gallons per day (mgd) and a daily average (for monthly reporting interval) of 5.6 mgd. Total residual halogen is limited to 0.1 mg/L or less and pH must be within the range of 6.5-9.0. The maximum daily effluent limit for chromium is 16.4 μg/L and the average monthly limit is 8.2 μg/L. Also, the maximum daily effluent limit for zinc is 107 μg/L and the average monthly limit is 53 μg/L. No discharge of polychlorinated biphenyl (PCB) compounds is permitted. The permit also limits the discharge of the 126 priority pollutants (40 CFR 423 Appendix A) contained in chemicals added for cooling tower maintenance, except chromium and zinc, to no detectable amount. Finally, the effluent limit for acute toxicity is no acute toxicity detected in a test concentration representing the acute critical effluent concentration (ACEC) (i.e., 11% effluent). Routine monitoring of these parameters assures that permit limits are not exceeded. Operation at the uprated conditions will not require modification of these permit conditions. The performance of the cooling towers has been evaluated under updated conditions, and it is determined that tower outlet temperature (and, therefore, blowdown temperature) will increase less than 1ºF. Analysis conducted during the previous NPDES permit cycle and field measurements conducted during operation support the judgment that a slightly warmer blowdown will not cause the water quality standard for temperature to be exceeded.

The state thermal discharge limits, the current discharges, and bounding analysis discharges for the TPO uprate are shown in Table 6-7. This comparison demonstrates that the plant remains within the state discharge limits during operation at TPO conditions.

6.4.3 Reactor Building Closed Cooling Water System

The heat loads on the Reactor Building closed cooling water (RCC) system do not increase significantly due to the TPO uprate. The main power-dependent heat loads on the RCC system that potentially would increase are those related to the operation of the RWCU non-regenerative heat exchangers, RWCU recirculation pumps, the reactor recirculation (RRC) pumps, drywell air coolers, and the fuel pool heat exchangers.

Changes to the RCC system heat loads are minimal (less than 1%) and will result in a negligible temperature increase (less than 1°F) for the RCC system during normal operation. The RCC system might experience a slight heat load increase (less than 1%) associated with the fuel pool coolers heat exchangers during refueling activities; however, the system has adequate design margin to remove any additional heat. Therefore, the RCC system is acceptable for the TPO uprate.

6.4.4 Ultimate Heat Sink

The UHS for CGS is the spray ponds. The SW system provides water from the UHS for equipment cooling throughout the plant. The ability of the UHS to perform the required safety functions at the 1.66% TPO uprate level is demonstrated with the UHS analyses based on 104.3% of CLTP. Therefore, all safety aspects of the UHS are within previous evaluations and the requirements are unchanged for the TPO uprate conditions. The current TS for UHS limits are adequate due to conservatism in the current 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 sub critical condition. The generic evaluation presented in TLTR Section 5.6.5 (SLCS) and Appendix L.3 (ATWS Evaluation) is applicable to the CGS TPO uprate. 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 SLCS relief valve margin is adequate for the TPO uprate because the SLCS system prior to the TPO uprate has a confirmed minimum relief valve margin of 132.5 psi (measured between the inlet to the SLCS relief valve and the minimum SLCS relief valve opening setpoint accounting for setpoint tolerance).

The SLCS ATWS performance is evaluated in Section 9.3.1. The evaluation shows that the TPO 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 heat load caused by the slightly higher FW process temperature (1 to 2°F). The increased heat load is within the margin of the steam tunnel air handling units. In the drywell, the increase in heat load due to the FW process temperature is within the system capacity. In the Turbine Building, the temperature increases are expected to be very low due to the increase in the FW process temperatures. Minor increases in the isophase bus duct heat rejection to the Turbine Building will be within the cooling capacity of the HVAC system. In the Reactor Building, there is no increase in heat load and the HVAC systems will continue to operate satisfactorily. TPO has no effect on the offgas charcoal adsorber vault HVAC system because this system has been abandoned in place and is no longer operational. Other areas (control room, Radwaste Building, and Emergency Diesel Generator Building) are unaffected by the TPO 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 and the potential for minor changes to combustible loading as a result of the TPO uprate are addressed by controlled design change procedures.

The operator manual actions that are being used for compliance with 10 CFR 50, Appendix R were reviewed. No operator manual actions have been identified in areas where environmental conditions, such as heat, would challenge the operator. 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 CGS 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. No “time-critical” repairs would be required to reach or maintain cold shutdown. The TPO and the additional decay heat removal would not affect the ability to reach and maintain cold shutdown within 72 hours.

Therefore, the fire protection systems and analyses are not affected by the TPO uprate.

6.7.1 10 CFR 50 Appendix R Fire Event

TLTR Section L.4 presents a generic evaluation of Appendix R events for an increase of 1.5% of CLTP. [[

]] The current analysis is based on 104.1% of CLTP and therefore establishes a bounding case for the clad temperature limit and the containment pressure limit.

Therefore, the generic results are applicable and no further plant-specific Appendix R analysis is necessary for the TPO uprate.

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. Other systems not addressed by this report are not significantly affected by TPO. The systems unaffected by TPO at CGS are confirmed to be consistent with the generic description provided in the TLTR.

Table 6-1 TPO Plant Electrical Characteristics

Parameter	Value
Generator Output (MWe)	1,227[(1)]
Rated Voltage (kV)	25
Power Factor	0.997
Generator Output (MVA)	1,230
Current Output (Amps)	28,406
Isolated Phase Bus Duct Rating: (Amps)	
Main Section	30,000
Delta Section	17,300
Auxiliary Section	1,200
Main Transformers Rating (MVA)	1,276 / 1,425

Note:

(1) Reactive power will be closely monitored at 1,227 MWe to ensure the 1,230 MVA rating of the main generator is not exceeded.

Table 6-2 Main Generator Ratings Comparison

Power Level	Design	Maximum Nominal	
	MVA @ 75 psig H_2	MWe @ 75 psig H_2	MVAR @ 75 psig H_2
Existing	1,230	1,206	273
Uprated[(1)]	1,230	1,227	50

Note:

(1) Operation at the uprated condition is not expected to have any adverse effect on the operation of the main generator. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures. Reactive power will be closely monitored at 1,227 MWe to ensure the 1,230 MVA rating of the main generator is not exceeded.

Table 6-3 Main Transformer Rating Comparison

Power Level	Design MVA at 65°C	MVA Loading
Existing	1,276 / 1,425	1,276 / 1,425
Uprated [(1)]	1,276 / 1,425	1,276 / 1,425

Note:

(1) Operation at the uprated condition is not expected to have any effect on the operation of the main transformer except if the spare main transformer is placed into service. The generator MWe will increase and MVAR will decrease, thus MVA will remain the same. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

Table 6-4 Normal Auxiliary Transformer Ratings Comparison

Winding Identification(1)	Rated MVA at 65°C	Existing MVA Loading	TPO MVA Loading
E-TR-N1/X – Winding	26.9	26.541	26.602
E-TR-N1/Y – Winding	17.9	12.050	12.079
E-TR-N2-X – Winding (2)	29.05	30.765	30.765

Notes:

(1) Operation at the uprated condition is not expected to have any effect on the operation of the normal auxiliary transformers excluding E-TR-N1 during the highest temperature months. There is a potential for plant downpower if transformer cooling cannot be maintained at 100% plant loading.

(2) Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

Table 6-5 Start-Up Auxiliary Power Transformer Comparison

Winding Identification[(1)]	Rated MVA at 65°C	Existing MVA Loading	TPO MVA Loading
E-TR-S/X - Winding	28.66	20.988	20.988
E-TR-S/Y – Winding	40	33.275	33.366

Note:

(1) Operation at the uprated condition is not expected to have any effect on the operation of the start-up auxiliary power transformer. Operation in this range is still within the operating boundaries specified in station design analysis and operating procedures.

Table 6-6 FPC System Parameters

Parameter	**CLTP**	**TPO**
Number of RHR/FPC trains	1 / 2	1 / 2
RHR heat exchanger flow rate, RHR/SW	3,500 / 7,200 gpm	3,500 / 7,200 gpm
Fuel pool heat exchanger flow rate, SFP/RCC	575 / 575 gpm	575 / 575 gpm
Design heat removal capacity (one RHR heat exchanger)	42.7E+6 BTU/hr	42.7E+6 BTU/hr
Design heat load, (2) fuel pool heat exchangers	8.00E+6 BTU/hr	8.00E+6 BTU/hr
Fuel cycle (months)	24	24
Bulk pool temperature (Normal Operation)	< 125°F	< 125°F
Bulk pool temperature (During Refueling)	< 150°F	< 150°F

Table 6-7 Effluent Discharge Comparison

Parameter	State Limit	Current	TPO
Flow (mgd)	5.6[1] and 9.4[2]	4.5[3]	Minimal change
Total Residual Halogen (mg/L)	0.1	< 0.1[4]	No change
Chromium, total (µg/L)	8.2[1] and 16.4[2]	< 1.8[5]	Minimal change
Zinc, total (µg/L)	53[1] and 107[2]	< 27[5]	Minimal change
PCBs	No discharge	No discharge	No change
The 126 priority pollutants (40 CFR 423 Appendix A) contained in chemicals added for cooling tower maintenance, except chromium and zinc	No detectable amount	No detectable amount	No change
pH	6.5-9.0	7.4-8.4[6]	No change
Acute Toxicity	No acute toxicity[7]	No acute toxicity[8]	Minimal change

Notes:

1. Average monthly effluent limit means the highest allowable average of daily discharges over a calendar month.
2. Maximum daily effluent limit is the highest allowable daily discharge.
3. Maximum daily average flow for September 2014 – August 2015.
4. Value verified by two samples collected at least 15 minutes apart prior to initiation of blowdown following biofouling treatments.
5. Maximum for September 2014 – August 2015.
6. Minimum and maximum pH range for September 2014 – August 2015.
7. No acute toxicity detected in a test concentration representing the ACEC. The ACEC equals 11% effluent.
8. Based on quarterly acute toxicity testing completed in 2015.

7.0 POWER CONVERSION SYSTEMS

7.1 TURBINE-GENERATOR

The CGS main T/G is designed with a maximum steam flow and generator capability in excess of rated conditions to ensure that the design rated output is achieved. The excess capacity ensures that the T/G 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. The difference in the steam flow capability between the current analyzed and rated conditions is called the flow margin.

The CGS T/G has a flow margin of 1.7% at the rated throttle steam flow of 14,127,313 lb/hr at a throttle pressure of 1,000 psia and rated electrical power output of 1,206 MW.

For the TPO uprate conditions of 3,544 MWt (approximately 101.66% of CLTP), the rated throttle steam flow is increased to 14,428,921 lb/hr at a throttle pressure of 999 psia. The evaluated increased throttle steam flow is approximately 101.66% of current rated steam flow. The evaluated increased throttle flow is due to the steam flow increase associated with operation at 101.66% CLTP conditions. The maximum uprated electrical output is expected to be 1,226 MW. Typical reactive power loading at the station is between -150 and +50 MVAR. Reactive power will be closely monitored at 1,226 MW to ensure the 1,230 MVA rating of the generator is not exceeded.

Steam specification calculations were performed to determine the TPO uprate turbine steam path conditions. These TPO uprate operating conditions are bounded by the previous analysis of the turbine and generator stationary and rotating components. Thus, the increased loadings, pressure drops, thrusts, stresses, overspeed capability and other design considerations resulting from operation at TPO RTP conditions are within existing design limits and operation; therefore, it is acceptable at the TPO uprate condition. In addition, valves, control systems and other support systems were evaluated, and TPO operating conditions are bounded by the existing analyses. The results of these evaluations show that no modifications are needed to support operation at the TPO uprate condition.

The existing rotor missile analysis was performed at 120% design overspeed conditions. The low-pressure turbine casing is designed to prevent rupture due to disc failure at 120% design overspeed conditions. The TPO uprate does not change turbine rated speed. Therefore, there is no change in the missile generation probability (a missile does not escape from the turbine casing) and thus, the missile generation probability remains unchanged and is therefore acceptable.

The overspeed evaluation addressed the sensitivity of the rotor train for the capability of overspeeding. Although the entrapped energy increases slightly for the TPO uprate conditions, no change in the overspeed trip settings is required because the existing analysis bounds the TPO uprate conditions.

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 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 currently operate at a steam flow capacity of approximately 23.75% of the 100% rated flow at CLTP. The steam bypass capacity at the TPO RTP is approximately 23.35% 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. The transient analyses that credit the turbine bypass system use a bypass capacity that is less than the actual capacity. Therefore, the turbine bypass capacity remains adequate for TPO operation because the actual capacity (unchanged) continues to bound the value used in the analyses.

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; however, their performance may affect the plant availability and capability to operate reliably at the TPO uprate condition.

A review of the CGS FW heaters, heater drain system, condensate demineralizers, and the pumps (condensate and FW) demonstrated that the components are capable of performing in the proper design range to provide the slightly higher TPO uprate FW flow rate at the desired temperature and pressure. A review of the CGS heater drain system demonstrated that the components will be 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 (RFWPs) 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 RFWP speed to increase the FW flow rate while still providing sufficient pressure at the RPV interface. Adequate margin during steady-state conditions also exists between the calculated minimum pump suction pressure and the low suction pressure trip setpoints.

The condensate and FW system functions adequately following a single RWFP trip in support of the NSSS to continue to operate without a reactor shutdown. Operation at the TPO condition continues to support this capability.

The existing RFW design pressure and temperature requirements bound operating conditions with adequate margin. The FW heaters are ASME Section VIII pressure vessels. The heaters were analyzed and 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.

Following a single FW pump trip, the RRS would runback recirculation flow such that the steam production rate is within the flow capacity of the remaining FW pump. However, the evaluation of a single FW pump trip event from rated or MELLLA core flow demonstrates that CGS will not avoid a reactor scram on low water level. Operation at the TPO condition does not change this conclusion.

7.4.3 Condensate Filters and Condensate Deep Bed Demineralizers

The effect of the TPO uprate on the condensate filter demineralizer (CFD) system (CPR) 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).

8.0 RADWASTE AND RADIATION SOURCES

8.1 LIQUID AND SOLID WASTE MANAGEMENT

The liquid radwaste system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse, discharge, or shipment.

Major sources of liquid and wet solid waste are from the CFDs. The TPO uprate results in an approximate 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; no increase is expected from the RWCU and FPC. 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 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. There are no significant environmental effects 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 offgas system design basis H_2 is 128 cfm (with a corresponding stoichiometric O_2 of 64 cfm). The expected H_2 flow rate for the TPO uprate is 106.7 cfm (53.4 cfm of O_2). The increase in H_2 and O_2 due to the TPO uprate remains well with the capacity of the system. Therefore, the TPO uprate does not 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 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 volt (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. They are expected to increase proportionally with power if the fuel irradiation time remains within the current basis. 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, the TPO uprate corrosion product concentrations are not expected to exceed the design basis concentrations. 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 CGS design. The total offgas rates for TPO uprate operations are bounded by the original design basis.

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 are expected to be 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. CGS was designed with substantial conservatism 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.

Post-operation 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 FPC system piping and 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 as low as reasonably achievable (ALARA) program, which controls access to radiation areas. The CGS radiation protection program procedural controls compensate for any minor increase in radiation levels due to the 1.66% 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. The slight increase in the post-accident radiation levels has no significant effect on the plant or 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

The TS limits implement the guidelines of 10 CFR 50, Appendix I. A review of the normal radiological effluent doses shows that at CLTP, the annual doses are a small fraction of the doses allowed by TS limits. 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

TLTR Appendix E provides a generic evaluation of the AOOs for TPO uprate plants. [[

]] Also included are the analytical methods to be used and operating conditions to be assumed. The AOO events are organized into two major groups: fuel thermal margin events and transient overpressure events.

TLTR Table E-2 illustrates the effect of a 1.5% power uprate on the OLMCPR. [[

]] The OLMCPR changes for the 1.66% uprate may be slightly larger than shown in Table E- 2, but the changes are expected to be within the normal cycle-to-cycle variation. The overpressure events and loss of FW transient are currently performed with the assumption of 2% overpower. Therefore, they are applicable and bounding for the TPO uprate.

The reload transient analysis includes the worst overpressure event, which is usually the closure of all MSIVs with high neutron flux scram.

The evaluations and conclusions of TLTR Appendix E are applicable to the CGS TPO uprate. Therefore, it is sufficient for the plant to perform the standard reload analyses at the first fuel cycle that implement the TPO uprate.

9.1.1 Alternate Shutdown Cooling Evaluation

Alternate shutdown cooling mode is not part of the CGS plant licensing basis.

9.2 DESIGN BASIS ACCIDENTS

The radiological consequences of a DBA are basically 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 or are not dependent on core thermal power. The Main Steam Line Break Accident 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 Regulatory Guides, the Standard Review Plan (SRP) (where applicable), and in previous SEs.

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transient Without Scram

CGS meets the following ATWS mitigation equipment requirements defined in 10 CFR 50.62:

1. Installation of an alternate rod insertion (ARI) system;
2. Boron injection equivalent to 86 gpm; and
3. Installation of automatic RPT logic (i.e., ATWS-RPT).

There are no changes in the equipment for the TPO uprate. The performance characteristics of the equipment do not change because operating conditions (operating pressure, SRV setpoints, and maximum rod line) do not change.

The CGS-specific analysis at the CLTP demonstrates that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1,500 psig;
2. PCT within the 10 CFR 50.46 limit of 2,200°F;
3. Peak clad oxidation within the requirements of 10 CFR 50.46;
4. Peak suppression pool temperature less than 212°F; and
5. Peak containment pressure less than 45 psig.

TLTR Section 5.3.5 and Appendix L present a generic evaluation of the sensitivity of an ATWS to a change in power typical of the TPO uprate. The evaluation is based on previous analyses for power uprate projects. For a TPO uprate, if a plant has sufficient margin for the projected changes in peak parameters given in TLTR Section L.3.5, then no plant-specific ATWS analysis is required. Sufficient margin is defined as having margin at least twice the generic adders available to the applicable plant limit prior to TPO. The generic TPO ATWS peak pressure and pool temperature adders are: (a) a 20-psi increase in RPV peak pressure for the MSIVC event and a 30-psi increase for the pressure regulator fail open (PRFO) event, and (b) a 1ºF increase in peak pool temperature for any limiting ATWS event. Therefore, the margins to the plant-specific limits considered sufficient before TPO are at least: (a) a 40-psi increase in RPV peak pressure for the MSIVC event and a 60-psi increase for the PRFO event, and (b) a 2ºF increase in peak pool temperature for any limiting ATWS event. The previous ATWS analysis, performed at 100% of CLTP, demonstrated a margin of 136 psi to the peak vessel bottom head pressure limit and a margin of 24°F to the pool temperature limit. These margins are in excess of the 60-psi and 2°F "sufficient margin" criteria defined in TLTR Appendix L. Therefore, no CGS-specific ATWS analysis is performed for the TPO uprate.

9.3.2 Station Blackout

The CGS plant station blackout (SBO) evaluation has previously been performed assuming ≥102% of CLTP. Therefore, the postulated SBO scenarios for TPO operation are bounded by the current evaluations.

10.0 OTHER EVALUATIONS

10.1 HIGH ENERGY LINE BREAK

Because the TPO uprate system operating temperatures and pressures change only slightly, there is no significant change in HELB mass and energy releases. These changes are insignificant in relation to the effect on line break calculations. Vessel dome pressure and other portions of the RCPB remain at current operating pressure or lower. Therefore, the consequences of any postulated HELB would not significantly change. The postulated break locations remain the same because the piping configuration does not change due to the TPO uprate.

The HELB evaluation was performed for all systems evaluated in the FSAR. 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 an 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. There is no increase in the steam flow calculated for a main steam line break accident (MSLBA). No change in the steam line break flow rate occurs because the flow restrictor and the operating pressure remain unchanged. The main steam line break (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.

10.1.2 Liquid Line Breaks

10.1.2.1 Feedwater Line Breaks

The TPO uprate increases the RFW temperature by less than 2°F and enthalpy by 1.3 BTU/lbm, which results in an insignificant increase in the RFW mass and energy release. As a result of the small increase in RFW energy, the blowdown rate changes marginally and the energy increases slightly. The MS tunnel HELB conditions are based on a MSLB in the tunnel; therefore, small changes in RFW process parameters have no effect on the MS tunnel HELB conditions. Therefore, the original HELB analysis is bounding.

10.1.2.2 ECCS Line Breaks

ECCS lines are normally isolated from the reactor during normal operations; therefore, the previous HELB analysis for breaks outside primary containment is bounding for the TPO uprate condition.

10.1.2.3 RCIC System Line Breaks

Because there is no increase in the reactor dome pressure relative to the original analysis, the mass flow rate does not increase. Therefore, the previous HELB analysis is bounding for the TPO uprate conditions.

10.1.2.4 RWCU System Line Breaks

As a result of the small decreases in RWCU process temperatures and enthalpies, the blowdown rate and energy released decrease slightly; therefore, the original HELB analyses bound the TPO uprate conditions.

10.1.2.5 CRD System Line Breaks

The CRD system and supporting equipment operation are not affected by a TPO uprate; therefore, the CRD pipe rupture analysis is not affected by the TPO uprate.

10.1.2.6 Building Heating and Auxiliary Steam Line Breaks

Building heating and auxiliary steam lines are not connected to the reactor-turbine primary loop. 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 Internal Flooding from HELB

None of the plant flooding zones contains a potential HELB location affected by the reactor operating conditions changed for the 1.66% 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 1.66% TPO uprate.

10.2 MODERATE ENERGY LINE BREAK

The plant flooding zones containing a potential moderate energy line crack (MELC) location are either unaffected or negligibly affected by the reactor operating conditions changed for the TPO uprate. The following systems contain potential MELC locations in plant flooding zones: condensate, SW, plant service water, RHR, reactor closed cooling water, demineralized water, fire protection, CRD, reactor core isolation cooling, low pressure core spray, FPC, standby liquid control and high pressure core spray.

No new moderate energy lines are identified from the 1.66% TPO uprate. Sources of moderate energy flooding and protection requirements for safe-shutdown equipment for a postulated MELC 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 RRC lines and will be evaluated through the Licensee Controlled Specification (LCS) 1.7.1 area 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 equipment outside containment are from a DBA-LOCA CLTP, the accident pressures, temperatures and radiation levels do not change.

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 100% of TPO RTP. 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 CGS.

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), 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. [[

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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. Operator actions for maintaining safe shutdown, core cooling, and containment cooling do not change for the TPO uprate. Minor changes to the P/F map and the flow-referenced setpoints will be communicated through normal operator training. Simulator changes and validation for the TPO uprate will be performed in accordance with established CGS plant simulator certification testing procedures.

10.6 PLANT LIFE

Two degradation mechanisms may be influenced by the TPO uprate: (1) irradiation assisted stress corrosion cracking (IASCC) and (2) FAC. 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 see a slight increase in fluence, but the inspection strategy used at CGS, 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.

The longevity of most equipment is not affected by the TPO uprate because there is no significant change in the operating conditions. No additional maintenance, inspection, testing, or surveillance procedures are required.

10.7 NRC AND INDUSTRY COMMUNICATIONS

NRC and industry communications are generically addressed in the TLTR, Section 10.8. Per 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

Plant procedures and programs are in place to:

1. Monitor and maintain instrument calibration during normal plant operation to assure 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 emergency operating procedures (EOPs) action thresholds are plant unique and will be addressed using standard procedure updating processes. It is expected that the TPO uprate will have a negligible effect or no effect on the operator action thresholds and to the EOPs in general.

10.10 INDIVIDUAL PLANT EXAMINATION

CGS maintains and regularly updates a station PRA model. Use of the model is integrated with station operations and decision-making.

The CGS IPE PRA model and analysis will not be specifically updated for TPO because the change in plant risk from the TPO uprate is insignificant because there is no change to plant operation, maintenance, or equipment design. This conclusion is supported by NRC RIS 2002-03 (Reference 4).

11.0 REFERENCES

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License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 12

Affidavits from Cameron (Caldon) Corporation Supporting the Withholding of Information in Enclosures 10 and 11 from Public Disclosure

Measurement Systems

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CAMERON

May 18, 2016
CAW 16-01

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1049 Rev. 3 "Bounding Uncertainty Analysis for Thermal Power Determination at Columbia Nuclear Generating Station Using the LEFM✓+M 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 16-01 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 16-01 and should be addressed to the undersigned.

Very truly yours,

Ernest M. Hauser
Director of Business Development

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

May 18, 2016
CAW 16-01

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

Sworn to and subscribed before me

this 18 day of

May, 2016

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 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:

   <u>Trade secrets and commercial information obtained from a person and privileged or confidential</u>

   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 are the submittals titled:

Engineering Report ER-1049 Rev. 3 "Bounding Uncertainty Analysis for Thermal Power Determination at Columbia Nuclear Generating Station Using the LEFM✓+M 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 M System used by Columbia Generating Station for flow measurement at the licensed reactor thermal power level of 3,544 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.

Measurement Systems

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CAMERON

May 18, 2016
CAW 16-02

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1074 Rev. 0 "Meter Factory Calculation and Accuracy Assessment for Columbia Nuclear Generating Station"

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 16-02 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 16-02 and should be addressed to the undersigned.

Very truly yours,

Ernest M. Hauser
Director of Business Development

Enclosures (Only upon separation of the enclosed confidential material should this letter and affidavit be released.)

May 18, 2016
CAW 16-02

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

Sworn to and subscribed before me

this 18 day of

May, 2016

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 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 are the submittals titled:
Cameron Engineering Report ER- 1074 Rev. 0 "Meter Factor Calculation and Accuracy Assessment for Columbia Nuclear Generating Station"

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 M System used by Columbia Nuclear Generating Station for flow measurement at the licensed reactor thermal power level of 3,544 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.

License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 13

Columbia Calculation NE-02-15-08, “Heat Balance Determination for Rated Thermal Power,” Revision 0

ENERGY NORTHWEST
DMS REFERENCE DOCUMENT INDEX SHEET

EC Number
14942

Primary Document Identification	AED	CAL	NE-02-15-08		0
	Document Type	Document Sub-Type	Document Number	Sheet Number	Document Revision

	Input References						Output References					
	ADD	DELETE	Type	Subtype	Doc Number	Sheet No	ADD	DELETE	Type	Subtype	Doc Number	Sheet No
1	☒	☐	PPI	PPM	9.3.1		☐	☐				
2	☒	☐	AED	EWD	EWD-14E-010B		☐	☐				
3	☒	☐	AED	EWD	EWD-14I-002		☐	☐				
4	☒	☐	AED	CAL	E/I-02-15-03		☐	☐				
5	☒	☐	AED	CAL	E/I-02-91-1138		☐	☐				
6	☒	☐	AED	CAL	E/I-02-15-05		☐	☐				
7	☒	☐	AED	CAL	E/I-02-15-06		☐	☐				
8	☒	☐	AED	CAL	E/I-02-15-07		☐	☐				
9	☒	☐	AED	CAL	E/I-02-15-08		☐	☐				
10	☒	☐	CVI	OMM	479-00,45		☐	☐				
11	☐	☐					☐	☐				
12	☐	☐					☐	☐				
13	☐	☐					☐	☐				
14	☐	☐					☐	☐				
15	☐	☐					☐	☐				
16	☐	☐					☐	☐				
17	☐	☐					☐	☐				
18	☐	☐					☐	☐				
19	☐	☐					☐	☐				
20	☐	☐					☐	☐				

Note:

Note:

ENERGY NORTHWEST CALCULATION	Calculation No. NE-02-15-08	
	Rev. No. 0	Page No. 1

Quality Class	Discipline	Status / F or S*	Initiating Documents
II	Nuclear	☒ F ☐ S	PDC 14942

Equipment Piece No.
Plant Process Computer (PPC)

*Study Calculations shall be used only for the purpose of evaluating alternate design options or assisting the engineer in performing assessments.

TITLE

Heat Balance Determination for Rated Thermal Power

PERFORMANCE/VERIFICATION RECORD

Prepared By: ESFL Qualified ☐ Yes ☒ No (Vendor 10 CFR Appendix B)

Print Name	Signature	Date
Bob Goff	Bob Goff	6-6-16

Verified By: ESFL Qualified ☐ Yes ☒ No (Vendor 10 CFR Appendix B)

Print Name	Signature	Date
Don Kinoshita	[signature]	6-6-16

Owner's Review By: ESFL Qualified ☐ Yes ☐ No

Print Name	Signature	Date

Approved By:

Print Name	Signature	Date

Unverified Assumption: ☐ Yes ☐ No If Yes, Resolution per AR# _____

Signature below denotes verification and resolution of unverified assumptions.

Print Name	Signature	Date

INCORPORATED ENGINEERING CHANGES (CMR, PDC, MALT, etc.)

Only list design changes if incorporated and shown as a current outstanding EC

N/A

REVISION NO.	REVISION SUMMARY
0	Initial Issue

TABLE OF CONTENTS

APPENDICES

OUTPUT INTERFACE DOCUMENT UPDATES

Affected Document No.:	Updated by:
N/A	
Deferred Changes Document No.:	**Tracked by:**
N/A	

Design Engineering Manager Approval for Deferred Changes:

N/A		
Print Name	Signature	Date

ENERGY NORTHWEST CALCULATION	Calculation No. NE-02-15-08	
	Rev. No. 0	Page No. 3

VERIFICATION CHECKLIST

Item:	YES	NO	N/A	Initials
Cover sheets properly completed.	☑	☐	☐	DK
Incorporated engineering change (EC) is complete and accurate.	☑	☐	☐	DK
Impacted output interface documents identified and updated.	☑	☐	☐	DK
Clear statement of purpose of analysis.	☑	☐	☐	DK
Methodology is clearly stated and appropriate for the proposed application.	☑	☐	☐	DK
Methodology justifies the use of engineer judgment.	☑	☐	☐	DK
Design inputs identified and adequately referenced to the source document.	☑	☐	☐	DK
Raw data (PDIS) used as design input are adequately validated.	☐	☑	☐	DK
Design criteria are suitable and properly referenced.	☑	☐	☐	DK
Reference documents list complete and sufficiently detail for document retrieval.	☑	☐	☐	DK
Analysis is logical, sufficiently detailed, arithmetically accurate, and specifies correct units.	☑	☐	☐	DK
Calculation results reasonable and correctly described in the Summary of Results and Conclusions.	☑	☐	☐	DK
Summary of Results and Conclusions includes discussion of margin.	☑	☐	☐	DK
Calculated values within the instrument display range.	☑	☐	☐	DK
Listed attachments included and complete.	☑	☐	☐	DK
Computer program identified with version and revision.	☐	☐	☑	N/A
Computer output included with program name, revision, run date on first page.	☐	☐	☑	N/A
Computer program verification/validation addressed.	☐	☐	☑	N/A
Native document files located in the appropriate EN file in accordance with DES-4-19.	☑	☐	☐	DK
	☐	☐	☐	
	☐	☐	☐	

Calculation Checking Method		Applicable pages/sections
Direct Step – By – Step Check	☑	
Alternate Calculation	☐	

Verified By: Calculation Verifier(s)

Print Name	Signature	Date
Don Kinoshita	[signature]	6-6-16

PURPOSE

Compute the Plant Process Computer (PPC) Reactor Thermal Power Core Heat Balance as part of the Measurement Uncertainty Recapture (MUR) Uprate.

SUMMARY OF RESULTS AND CONCLUSIONS

The Measurement Uncertainty Recapture allows a licensed power level that maintains margin to 102% of Current Licensed Thermal Power (CLTP). The CLTP is 3486 MW. 102% of CLTP is:

102/100*3486 = 3556 MW

The proposed LTP plus the heat balance measurement uncertainty (U_{RTP}) must remain ≤ 3556 MW or:

Proposed LTP ≤ 3556 MW - U_{RTP}

For LEFM in ✓+ mode:

Proposed LTP ≤ 3556 MW - 11.649 MW = 3544.351 MW
Proposed LTP (Rounded off) = 3544 MW

For LEFM in check (maintenance)mode:

Proposed LTP ≤ 3556 MW - 18.586 MW = 3537.414 MW
Proposed LTP (Rounded off) = 3537 MW

METHODOLOGY

☒ Manual (As required, document source of equations in Reference List)

☐ Verified Program: Software/Revision _____

☐ Alternate Verification Documented in Appendix _____

The method used in performing this calculation is based on Supply System Standard EES-4 “Setpoint Methodology” which is based on ANSI/ISA-S67.04-1988, “Setpoints for Nuclear Safety-Related Instrumentation: and guidelines in ISA draft Recommended Practice RP67.04 “Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation”. This method is adapted below to only determine loop uncertainty. This calculation does not determine a setpoint.

Note: For a definition of terms, refer to EES-4.

1. Define the loop to be analyzed.
2. Determine the normal environmental conditions for each component of the loop. (accident conditions are not considered per this calculation)
3. Determine the normal environmental effects on the accuracy of each instrument in the loop. This includes, but is not limited to the following effects:

   Pressure, temperature, humidity, radiation, seismic.

4. In addition to the above environmental effects the following accuracy effect must also be evaluated if applicable:

   Power supply effects, process effects, reference accuracy, static pressure effects, primary element effect, setting accuracy effects, insulation resistance effect, measuring and test equipment uncertainty effects, any additional loop specific effects

5. Determine the normal drift effects for each instrument in the loop using plant data if possible or vendor data or the default value of 1% of equipment span or setting per calibration period.

6. Combine the effect terms as defined in EES-4

DESIGN INPUTS

6. **Instrument Loop Operational Characteristics**

   6.1. **Description of Instrument Loop**

System No.: 14

EWD: EWD-14E-010B, EWD-14I-002

Equipment Description: This Plant Process Computer (PPC) loop calculates and displays reactor thermal power (Heat Balance) using current plant data from multiple inputs.

Loop Function

Normal: The Heat Balance is a process of defining a control boundary around the reactor vessel and subtracting all of the energy flowing inward through the boundary from all of the energy exiting outward from the boundary. The net difference is the power produced within the boundary which is defined as the Core Thermal Power. The Heat Balance Equation is as follows:

CTP = QFW + QCR + QCU + QRAD – QPUMP

Where: CTP = Core Thermal Power in Megawatts (MW).
QFW = QFW =Net energy of the Feed Water System (MW).
QCR = QCR =Net energy of the Control Rod Drive System (MW).
QCU = QCU =Net energy of the Reactor Water Cleanup System (MW).
QRAD = QRAD =Net energy of radiative sources (MW).
QPUMP = QPUMP =Net energy of the Reactor Recirculation Pumps (MW).

The parameter with the most significant impact on the heat balance is the net energy from the Feed Water system (QFW). The Feed Water Flow measurement accounts for more than 99.5% of the calculated CTP at normal full power conditions. Therefore, it is desired to have the most accurate available data source when determining the Feed Water Flow. The next most important parameters are the Feed Water Temperature and the Reactor Pressure.

The following computer points/instrument loops are used as inputs to the PPC Heat Balance calculation:

LEFM 206 Feed Water Flow Rate Loop A, mlb/hr
LEFM 222 Feed Water Flow Rate Loop B, mlb/hr
LEFM 210 Feed Water Temperature Loop A, degf
LEFM 226 Feed Water Temperature Loop B, degf
X136 Narrow Range MS-PT-808 – Reactor Pressure, psig – preferred

6.1. **Description of Instrument Loop**

X054 Wide Range MS-PT-51B – Reactor Pressure, psig
X151 Wide Range MS-PT-51A – Reactor Pressure, psig
B020 CRD Flow Rate CRD-FT-4, mlb/hr
B021 RWCU System Flow Rate RWCU-FT-89A, (Filter Demin A), mlb/hr
B024 RWCU System Flow Rate RWCU-FT-89B, (Filter Demin B), mlb/hr
B047 RWCU System Temperature Leaving Reactor RWCU-TE-4, degf
B048 RWCU System Temperature Entering Reactor RWCU-TE-15, degf
B031 Recirculation Pump A Power, RRC-P-1A, MW
B032 Recirculation Pump B Power, RRC-P-1B, MW

Basis: Refs. 1, 3.

Accident: N/A

Basis: N/A

Specific Calculation Information:

Assumptions:

1. Uncertainty due to PPC input signal processing and A/D conversion is included in the uncertainty calculations for each computer point value used in the PPC Heat Balance.
2. Uncertainty resulting from approximations used in the PPC Heat Balance calculation software is negligible.
3. The CRD temperature is assumed to be a constant 80 °F. This assumption is conservative because it is below operating temperature and causes only a small error.
4. QRAD is assumed to be -1.1.MW. Per Reference 1, uncertainty is negligible.
5. Steam quality is assumed to be 1. This assumption is conservative.
6. FW and RWCU (compressed water) enthalpy uncertainties due to reactor pressure uncertainty are negligible.
7. Uncertainty for assumed RRP efficiency of 94% is negligible.
8. Uncertainty for barometric pressure effect on conservative steam pressure uncertainty of ±15 psi is negligible.

6.2 **Instrument Description and Uncertainties for PPC Heat Balance**

Description

Manufacturer:	Fairchild Weston System Inc.
Model:	N/A
Other Description:	PPC Replacement System
Instrument Type:	Computer
Basis:	Ref. 16

	Low	High	Units	Basis
Instrument Range Calibrated Span:	10	>100	% RTP	Ref. 1,3
Input:	N/A	N/A	Computer Point Values	Ref. 1,3
Output:	0	>3544	MWT	Ref. 1,3

6.2 **Instrument Description and Uncertainties for PPC Heat Balance**

Environmental Conditions

	Calibration	Normal Min.	Normal Max.	DBE (peak)	Basis
Temperature (°F)	N/A	N/A	N/A	N/A	Ambient temperature, humidity, and pressure have no impact on the digital PPC data.
Pressure (psia)	N/A	N/A	N/A	N/A	See above
Humidity (%RH)	N/A	N/A	N/A	N/A	See Above
Process T (°F)	N/A	N/A	N/A	N/A	Process temperature, and pressure have no impact on the digital PPC data.
Process P (psig)	N/A	N/A	N/A	N/A	See Above
Radiation (*)	N/A	N/A	N/A	N/A	Not susceptible to radiation, (See RE)

*Normal Max. = TID rads for 40 years * 1,000/40 yrs/365.25 days/24 hrs = mr/hr
DBE = dose rate for period during which instrument must operate (rad/hr)

Calibration Period (t_c) (months)	N/A	Basis: The PPC computer points used for the Heat Balance are calibrated with their associated instrument loops.
Power Supply Stability (PSS) (V)	N/A	Basis: The PPC digital data is not affected by PSS.

RA – Reference Accuracy

RA = 0% of CS Basis: The PPC computer points used for the Heat Balance are calibrated with their associated instrument loops.

M&TE – Measurement and Test Equipment Uncertainty

M&TE = 0% of CS Basis: The PPC computer points used for the Heat Balance are calibrated with their associated instrument loops.

SA – Setting Accuracy

SA = I + o

SA = 0 + 0

SA = 0% of CS

Where:		Basis:
i = Reading Accuracy of Input	0 % of CS	The PPC computer points used for the Heat Balance are calibrated with their associated instrument loops.
o = Setting Accuracy of Output	0 % of CS	Output is digital display.

6.2 **Instrument Description and Uncertainties for PPC Heat Balance**

CAL – Calibration Effect

$\mathrm{CAL} = \sqrt{M\&TE^2 + SA^2}$
$\mathrm{CAL} = \sqrt{0^2 + 0^2}$
$\mathrm{CAL} = 0\%$ of CS

DR – Drift

DR^+ = 0% of CS Basis: Assumptions 2, 3.
DR^- = 0% of CS Basis: See above.

HE – Humidity Effect

HE = 0% of CS Basis: The PPC digital data is not affected by humidity.

IR – Insulation Resistance Effect

IR = 0% of CS Basis: The PPC digital data is not affected by IR.

PSE – Power Supply Effect

PSE = 0% of CS Basis: The PPC digital data is not affected by PSE.

PE – Pressure Effect

PE = 0% of CS Basis: The PPC digital data is not affected by pressure.

PPE – Process/Primary Element Effect

PPE^+ = 0% of CS Basis: Assumption 3.

PPE^- = 0% of CS Basis: See above.

RE – Radiation Effect

RE = 0% of CS Basis: The PPC digital data is not affected by radiation levels in the control room.

SE – Seismic Effect

SE = 0% of CS Basis: The PPC is seismic category 2 and is not qualified under SSE loading conditions.

SP – Static Pressure Effect

SP = 0% of CS Basis: The PPC digital data is not affected by pressure.

TE – Temperature Effect

TE = 0% of CS Basis: The PPC digital data is not affected by temperature.

6.2 **Instrument Description and Uncertainties for PPC Heat Balance**

<u>Input Loop Uncertainties</u>

Each PPC computer point used as an input to the Heat Balance has a calculated loop uncertainty as follows:

PPC Point	Process Input	Uncertainty	Units	Basis Document/ Ref. #
LEFM 206	Feed Water Flow Rate Loop A ✓+	±0.239 Combined	%	ER-1049 Rev. 3 (Note 1)/ 2
LEFM 222	Feed Water Flow Rate Loop B.✓+		%	ER-1049 Rev. 3 (Note 1)/ 2
LEFM 206	Feed Water Flow Rate Loop A ✓	±0.463 Combined	%	ER-1049 Rev. 3 (Note 1)/ 2
LEFM 222	Feed Water Flow Rate Loop B ✓		%	ER-1049 Rev. 3 (Note 1)/ 2
LEFM 210	Feed Water Temperature Loop A	±0.6 Combined	degf	ER-1049 Rev. 3 (Note 1)/ 2
LEFM 226	Feed Water Temperature Loop B		degf	ER-1049 Rev. 3 (Note 1)/ 2
X136	MS-PT-808 – Reactor Pressure	±2.00	psi	E/I-02-15-03 Rev. 0/ 10
X054	MS-PT-51B – Reactor Pressure	±14.85	psi	E/I-02-91-1138 Rev. 0/ 11
X151	MS-PT-51A – Reactor Pressure	±14.85	psi	E/I-02-91-1138 Rev. 0/ 11
B020	CRD Flow Rate CRD-FT-4 Random	+0.0016254 -0.0017185	mlb/hr	E/I-02-15-05 Rev. 0/ 12
B020	CRD Flow Rate CRD-FT-4 Non-Random	+0.0004556 -0.0004626	mlb/hr	E/I-02-15-05 Rev. 0/ 12
B021	RWCU System Flow Rate RWCU-FT-89A Filter Demin A Random	+0.00698 -0.00791	mlb/hr	E/I-02-15-06 Rev. 0/ 13
B021	RWCU System Flow Rate RWCU-FT-89A Filter Demin A Non-Random	+0.00290 -0.00305	mlb/hr	E/I-02-15-06 Rev. 0/ 13
B024	RWCU System Flow Rate RWCU-FT-89B Filter Demin B Random	+0.00698 -0.0791	mlb/hr	E/I-02-15-06 Rev. 0/ 13
B024	RWCU System Flow Rate RWCU-FT-89B Filter Demin B Non-Random	+0.00290 -0.00305	mlb/hr	E/I-02-15-06 Rev. 0/ 13
B047	RWCU System Temperature Leaving Reactor RWCU-TE-4	±7.44	degf	E/I-02-15-07 Rev. 0/ 14
B048	RWCU System Temperature Entering Reactor RWCU-TE-15	±7.44	degf	E/I-02-15-07 Rev. 0/ 14
B031	Recirculation Pump A Power, RRC-P-1A, Random	±0.1026	MW	E/I-02-15-08 Rev. 0/ 15
B031	Recirculation Pump A Power, RRC-P-1A, Non-Random	±0.3372	MW	E/I-02-15-08 Rev. 0/ 15
B032	Recirculation Pump B Power, RRC-P-1B, Random	±0.1026	MW	E/I-02-15-08 Rev. 0/ 15
B032	Recirculation Pump B Power, RRC-P-1B, Non-Random	±0.3372	MW	E/I-02-15-08 Rev. 0/ 15

<u>Note1:</u> A conservative LEFM total error contribution of ±0.3% RTP is used (in lieu of Feed Water Flow and Temperature uncertainties) for ✓+ mode, ±0.5% RTP is used for check (maintenance) mode.

Reference Document List

Reference Number	Author	Issue Date, Edition or Revision	Title	Document Number
1	General Electric	Rev. 2	Reactor Heat Balance	T0100
2	*Caldon Ultrasonics*	*Rev. 3*	*Bounding Uncertainty Analysis for Thermal Power Determination at Columbia Nuclear Generating Station Using the LEFM✓+ System*	*ER-1049*
3	Energy Northwest	Rev. 15	Manual Core Heat Balance	9.3.1
4	Energy Northwest	Rev. 7-1	Setpoint Methodology	PPI EES-4
5	Energy Northwest	Rev. 5	Electrical Wiring Diagram for Process Computer System	EWD-14E-010B
6	Energy Northwest	Rev. 7	Electrical Wiring Diagram Process Computer System Interface	EWD-14I-002
7	Energy Northwest	Amend 228	Technical Specifications	1.1
8	ASME	1967	Steam Tables	N/A
9	Energy Northwest	12	Feedwater Flow & Temperature Measurement Upgrade	EC 7855
10	Energy Northwest	Rev. 0	Instrument Loop uncertainty determination for MS-PT-808 to PPC	E/I-02-15-03
11	Energy Northwest	Rev. 0	Setpoint Determination for Instrument Loops MS-PT-51A and MS-LR/PR-623A	E/I-02-91-1138 CMR 15010
12	Energy Northwest	Rev. 0	Instrument Loop Uncertainty Determination for CRD-FT-4	E/I-02-15-05
13	Energy Northwest	Rev. 0	Instrument Loop Uncertainty Determination for RWCU-FT-89A & 89B	E/I-02-15-06
14	Energy Northwest	Rev. 0	Determination of Uncertainty Calculation for Instrument Loop RWCU-TE-4 and RWCU-TE-15	E/I-02-15-07
15	Energy Northwest	Rev. 0	Instrument Loop Uncertainty Determination for RRC-WTD-RRA & RRB	E/I-02-15-08
16	Fairchild Weston Systems Inc.	Rev. 0	Instruction Manual for Plant Process Computer Replacement System	CVI-479-00-45

BODY OF CALCULATION - ANALYSIS

6.3 **Instrument Loop Uncertainties for PPC Heat Balance**

U_D – Device Uncertainty

In order to calculate total uncertainty for the PPC Heat Balance indication at or near the Reactor Heat Balance Task Report T0100 RTP target value (3544 MW), the individual contribution to output uncertainty (in MW) must be calculated for each input loop uncertainty. The conversion to MW uncertainty is performed by applying the instrument loop uncertainty relative to 3544 MW and the process conditions specified in T0100 Figure 3-1. MW uncertainty decreases with decreasing RTP, therefore using 3544 MW is conservative for lower power.

Feed Water

A conservative LEFM total error contribution of ±0.3% RTP is used (in lieu of Feed Water Flow and Temperature uncertainties) for ✓+ mode, ±0.5% RTP is used for check (maintenance) mode. These uncertainties bound the results in ER-1049 Rev. 3.

U_{FW} = ±0.3% RTP = ±0.3%*3544 MW = ±10.632 MW for ✓+ mode

U_{FW} = ±0.5% RTP = ±0.5%*3544 MW = ±17.720 MW for check (maintenance) mode

Reactor Pressure

Reactor pressure uncertainty used to bound the available instrument loops = ±15 psi

-15 psi is used for bounding enthalpy uncertainty

T0100 Figure 3-1 enthalpy of 1191.6 btu/lbm @1035 psia is raised to 1192.2 btu/lbm @1020 psia

U_{RP} MW = (∆enthalpy btu/lbm)*(steam flow rate mlbm/hr)*(17.584 watts/BTU/min)*(1hr/60 min)

U_{RP} MW = (1192.2-1191.6)*(15.2890)*(17.584 watts/BTU/min)*(1hr/60 min) = ±2.6884 MW

CRD System Flow

Per Ref. 3, 14 gpm is added to measured value to account for bypass (unmonitored) CRD flow. This nominal addition is conservative, and is not considered for uncertainty.

CRD System Flow uncertainty is +0.0016254, -0.0017185 mlbm/hr random, +0.0004556, -0.0004626 mlbm/hr non-random

-0.0017185 mlbm/hr random and -0.0004626 mlbm/hr non-random are used as bounding flow uncertainty

T0100 Figure 3-1 CRD enthalpy of 50.9 btu/lbm and saturated steam enthalpy of 1191.6 btu/lbm @1035 psia are used to determine ∆enthalpy

U_{CRD} MW = (∆enthalpy btu/lbm)*(CRD flow rate error mlbm/hr)*(17.584 watts/BTU/min)*(1hr/60 min)

6.3 **Instrument Loop Uncertainties for PPC Heat Balance**

U_{CRDR} MW = (1191.6-50.9)*(0.0017185)*(17.584 watts/BTU/min)*(1hr/60 min) = ±0.57450 MW

U_{CRDB} MW = (1191.6-50.9)*(0.0004626)*(17.584 watts/BTU/min)*(1hr/60 min) = ±0.15465 MW

RWCU System Flow

RWCU System Flow uncertainty per loop is +0.00698, -0.00791 mlbm/hr random, +0.00290, -0.00305 mlbm/hr non-random

-0.00791 mlbm/hr random, and -0.00305 mlbm/hr non-random are used as bounding flow uncertainty per loop

T0100 Figure 3-1 RWCU inlet enthalpy of 528.3 btu/lbm and outlet enthalpy of 386.1 btu/lbm are used to determine ∆enthalpy

U_{RWCUF} MW = (∆enthalpy btu/lbm)*(RWCU flow rate error mlbm/hr)*(17.584 watts/BTU/min)*(1hr/60 min)

U_{RWCUFR} MW = (528.3-386.1)*(0.00791)*(17.584 watts/BTU/min)*(1hr/60 min) = ±0.3296 MW

U_{RWCUFB} MW = (528.3-386.1)*(0.00305)*(17.584 watts/BTU/min)*(1hr/60 min) = ±0.1271 MW

RWCU Inlet and Outlet Temperature

RWCU System Temperature uncertainty is ±7.44 degf for each temperature

$\sqrt{7.44^2 + 7.44^2}$ = ±10.52 degf is used as bounding temperature uncertainty (PPC non-random drift does not affect ∆t)

+10.52 degf applied to RWCU inlet is used for bounding enthalpy uncertainty

T0100 Figure 3-1 enthalpy of 528.3 btu/lbm @533.4 degf is raised to 541.5 btu/lbm @543.92 degf to determine ∆enthalpy

U_{RWCUT} MW = (∆enthalpy error btu/lbm)*(RWCU flow rate mlbm/hr)*(17.584 watts/BTU/min)*(1hr/60 min)

U_{RWCUT} MW = (541.5-528.3)*(0.1813)*(17.584 watts/BTU/min)*(1hr/60 min) = ±0.70136 MW

Recirculation Pump Power

Recirculation Pump Power indication uncertainty per pump is ±0.1026 MW random, ±0. 3372 MW non-random

Recirculation Pump efficiency used for heat balance is 94% per Ref 3.

U_{RPP} heat input uncertainty MW = power input uncertainty MW *0.94

U_{RPPR} MW = ±0.1026 MW*0.94 = 0.09644 MW per pump

U_{RPPB} MW = ±0.3372 MW*0.94 = 0.3170 MW per pump

6.4 **Total Loop Uncertainty**

Total Loop Uncertainty is obtained by combining the input loop uncertainties per EES-4:

Random uncertainties are combined by SRSS methodology.

The non-random uncertainties are dominated by PPC A/D drift (Refer to Appendix A). This cyclic A/D drift is common to affected heat balance inputs. For A/D drift in the same direction, the non-random uncertainties for CRD/RWCU flows and recirculation pumps have opposing effects on the calculated CTP. This is because the QPUMP term in the heat balance equation is negative:

CTP = QFW + QCR + QCU + QRAD – QPUMP

Since the non-random uncertainty of QPUMP (U_{RPPB}) is larger than the opposing non-random flow uncertainties (i.e., bounding), it is added to the SRSS result.

For LEFM in ✓+ mode:

$$U_{RTP} = \pm K\left[\sqrt{\left(U_{FW}{}^{2} + U_{RP}{}^{2} + U_{CRDR} + U_{RWCUFR}{}^{2} + U_{RWCUT}{}^{2} + U_{RPP}{}^{2}\right)} \pm U_{RPPB}\right]$$

$$U_{RTP} = \pm 1\left[\sqrt{(10.632^2 + 2.6884^2 + 0.57450^2 + 0.3296^2 + 0.3296^2 + 0.70136^2 + 0.09644^2 + 0.09644^2)} \pm 2 * (0.3170)\right]$$

$$U_{RTP} = \pm 11.649\ MW$$

For LEFM in check (maintenance)mode:

$$U_{RTP} = \pm K\left[\sqrt{\left(U_{FW}{}^{2} + U_{RP}{}^{2} + U_{CRDR} + U_{RWCUFR}{}^{2} + U_{RWCUT}{}^{2} + U_{RPP}{}^{2}\right)} \pm U_{RPPB}\right]$$

$$U_{RTP} = \pm 1\left[\sqrt{(17.720^2 + 2.6884^2 + 0.57450^2 + 0.3296^2 + 0.3296^2 + 0.70136^2 + 0.09644^2 + 0.09644^2)} \pm 2 * (0.3170)\right]$$

$$U_{RTP} = \pm 18.586\ MW$$

6.4 **Total Loop Uncertainty**

<u>Summary for Instrument Loop</u>

The Measurement Uncertainty Recapture allows a licensed power level that maintains margin to 102% of Current Licensed Thermal Power (CLTP). The CLTP is 3486 MW. 102% of CLTP is:

102/100*3486 = 3556 MW

The proposed LTP plus heat balance measurement uncertainty must remain ≤ 3556 MW or:

Proposed LTP ≤ 3556 MW - U_{RTP}

For LEFM in ✓+ mode:

Proposed LTP ≤ 3556 MW - 11.649 MW = 3544.351 MW

Proposed LTP (Rounded off) = 3544 MW

For LEFM in check (maintenance)mode:

Proposed LTP ≤ 3556 MW - 18.586 MW = 3537.414 MW

Proposed LTP (Rounded off) = 3537 MW

B-PPC channel non-random drift 95/95 statistical analysis

Preparer: Ralph Berger *Ralph Berger* Date: 5/25/2016

Reviewer: Don Kinoshita *Don Kinoshita* Date: 5/25/2016

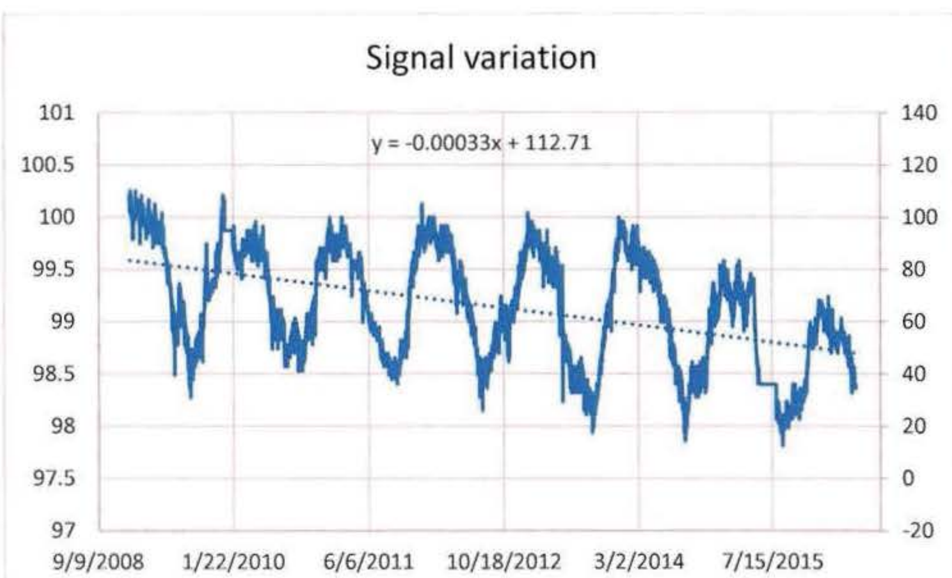
Purpose and Results

This section develops the uncertainty in the PPC term as a non-random drift of 1.81% to be applied as +1.81%/-1.81%. A 95/95 methodology is utilized from ISA RP67.04.02-2000 to provide 95% confidence that this range will address 95% of the variation. Further conservatism is added by enveloping all sources of uncertainty into a non-random drift and applying it in the most limiting direction. The computer points that have this non-random drift are associated with analogue Columbia GS B-PPC points.

Input Data

The test signals for PPC reference B019 were transmitted by memo, Steele to Menocal, May 24, 2016, and consist of 64,668 data obtained from Plant Process Computer point B019MV via the "eDNA" historian for the dates 1 -1 -2009 to 5-16-2016 (see AR 344042-69). Of these data, 62,277 are labeled "OK" for use. The data consist of output voltages given an input signal of 100 mV. A plot of the data is shown below where the roughly 2,400 bad data are replaced with the previous good data point for visual clarity.

The signal evidences a sinusoidal error with a slow drift of about 0.11% per year (calculated by the linear trend line fit, which gives .00033%/day). The sinusoidal pattern correlates very well with annual temperature variation, so it is a long-term impact that is inconsistent with random fluctuations as random errors are described in procedure EES-4. It is conservative to treat the full signal variation (peak to trough) as a non-random drift (or systematic uncertainty), since it will add to the total uncertainty in full in the limiting direction, instead of being a random factor within the SRSS radical. Note that the final result incorporates the annual temperature effects, the slow drift, and the random noise in a single term.



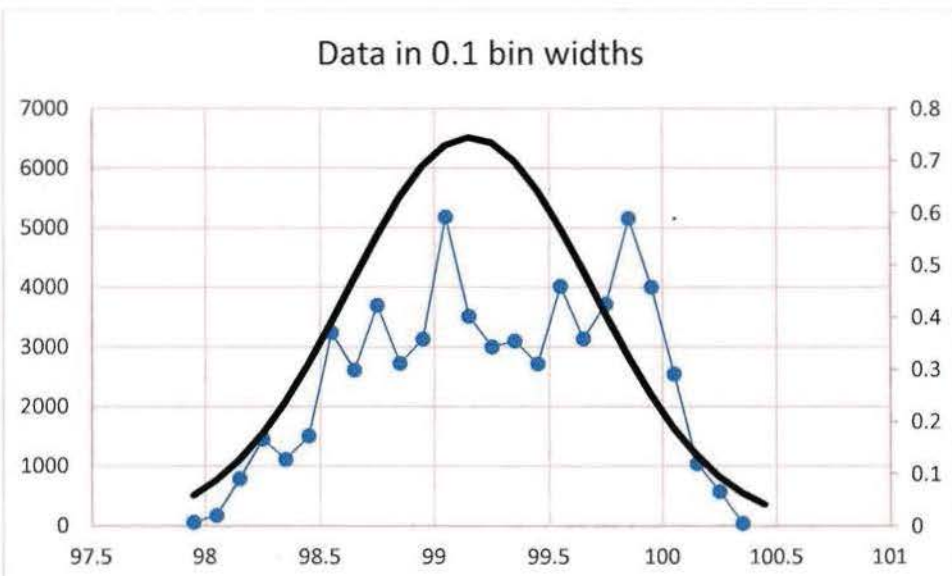
Assumptions

1. The example drift from point B019 can be applied to all other B-PPC channels.

Methodology and Body of Calculation

The 95/95 value for peak to trough variation is developed using Annex E to ISA RP67.04.02-2000. Since calibration is performed on a two year cycle, the variation is observed in rolling periods of two years (specifically, the six overlapping periods of 1/1/2009 to 1/1/2011; 1/1/2010 to 1/1/2012; 1/1/2011 to 1/1/2013; 1/1/2012 to 1/1/2014; 1/1/2013 to 1/1/2015; and 1/1/2014 to 1/1/2016). The worst 95/95 limits for those periods is discovered, which will be seen to be 1/1/2013 to 1/1/2015, with values of 99.916 and 98.106, making a total peak to trough uncertainty of 1.81%.

The methodology for a non-normally distributed data set is used. That the data is not a normal distribution can be determined in a variety of ways, one of which is to view the above data graph. This is not a signal that is centered on a mean, but rather a signal that drifts back and forth from a high to low region. A Kurtosis test result is +3 if perfectly bell-shaped, but our results are around -1, meaning fairly flat. The full data set is binned into 0.1 widths and plotted below, with a normal curve based on the data set mean and standard deviation for comparison. It is visually clear that the normal distribution is not a good match.



For the normal curve above, the mean is 99.16 and the standard deviation is 0.5356 (Excel function average() and stdev() applied to all good data points). This implies a 2.5% and 97.5% value of 99.16 +/- 1.96*.5356 = 98.11, 100.21, or a 2.10% peak to trough value. This is higher than the final result below in part because it includes 7 years of drift (instead of our two-year period between calibration), but it still provides a reasonable order of magnitude check for our final result of 1.81%.

The ISA 67.04.02-2000 Annex E methodology for the 95/95 limits of a non-normal data set are presented in Section E.4.2 using a minimum pass probability model, which is also described in Beggs, W.J., Statistics for Nuclear Engineers and Scientists, Part 1: Basic Statistical Inference," DOE Research and Development Report No. WAPD-TM-1292, February, 1981. An example on page 94 of Beggs verifies the approach.

Equation E-4 of ISA 67.04.02-2000 is:

$$P_{u,1} = x/n \pm z\{(1/n)(x/n)[1 - (x/n)]\}^{1/2} \quad \text{[E-4]}$$

For our purposes, $P_{u,1}$ is either 2.5% or 97.5%, which determines the limits for the first 95 in our 95/95 confidence. N is our sample size, which for two years is on the order of 8760 hours of data*2 = 17520 data. Z is the confidence interval from a normal curve, which is 1.96 for our second 95 in the 95/95 confidence description. x/n is a term that is developed from the equation by iteration. It is defined as the fraction of data "outside the pass/fail criterion." We will be using x/n to determine the range at which we can say, with 95/95 confidence, that we have reached the 2.5% or 97.5% value.

By examination of Equation E-4, we see that if the sample size was infinite (1/n = 0), then $P_{u,1}$ would equal x/n. That is, the 2.5% 95/95 value would be the 2.5% largest value when ranking all values from minimum to maximum. Similarly, with an infinite sample size, the 97.5% value would be the 97.5% term in our ranked list of data. With our sample size of 1/n = 1/17,500, it takes a x/n value on the order of 2.3% to produce a value of $P_{u,1}$ = 2.5%, and x/n = 97.7% to produce a $P_{u,1}$ value of 97.5%. In other words, when we rank our data, we can say with 95% confidence that our observed range from 2.3% to 97.7% will encompass all future data.

Mathematically, the formula for our first period, which contains 17,404 data, is:

$$0.025 = P_{u,2.5} = x/n \pm z\{(1/n)(x/n)[1 - (x/n)]\}^{1/2} = 0.022783 + 1.96*(1/17404*.022783*(1-.022783))^{0.5}$$

Here the value of .022783 was found by iteration, and is a function of sample size only. Similarly, the upper limit is found from

$$0.975 = P_{u,97.5} = x/n \pm z\{(1/n)(x/n)[1 - (x/n)]\}^{1/2} = 0.977217 - 1.96*(1/17404*.977217*(1-.977217))^{0.5}$$

The process to determine the 95/95 limits is to first take the full seven years of data, 64,668 items, and arrange in cells D3:D64670 in an Excel worksheet PPC. Column C is the designation of "OK" or "Unreliable" or "Unavailable." The top of our spreadsheet of data appears like this:

	A	B	C	D
1				
2	FEEDFLOWCALIBRATIONST			
3	1/1/2009	0:00:00	OK	100.1683
4	1/1/2009	1:00:00	OK	100.1683
5	1/1/2009	2:00:00	OK	100.2104
6	1/1/2009	3:00:00	OK	100.1683
7	1/1/2009	4:00:00	OK	100.2104
8	1/1/2009	5:00:00	OK	100.2104
9	1/1/2009	6:00:00	OK	100.1683
10	1/1/2009	7:00:00	OK	100.1683
11	1/1/2009	8:00:00	OK	100.1683
12	1/1/2009	9:00:00	OK	100.1683
13	1/1/2009	10:00:00	OK	100.1683

Our six periods of data are characterized as follows, with the solution of Equation E-4 in the right two columns.

Period	Two year periods (1/1 to 1/1)	Rows	n	$P_{u,2.5\%}$	$P_{u,97.5\%}$
1	2009-2011	3 to 17523	17404	0.022783	0.977217
2	2010-2012	8763 to 26283	17237	0.022773	0.977227
3	2011-2013	17523 to 35067	17264	0.022775	0.977225
4	2012-2014	26283 to 43827	17465	0.022787	0.977213
5	2013-2015	35067 to 52587	17425	0.022784	0.977216
6	2014-2016	43827 to 61347	15726	0.022673	0.977327

Here the sample size n is calculated by typical formula =COUNTIF(PPC!C3:C17523,"OK"). The 6th period has bad data in May to July of 2015 causing the low value of n. The total number of rows per period is either 2*8760 = 17520, or 17544 for the periods 3 and 4 that include the leap year 2012.

Next, the good data is collected in cells k3:k17523 in spreadsheet ANALYSIS with equations such as =IF(PPC!C3="OK",PPC!D3,""), replicated to capture all of the good data in the first two years. Those results are copied, pasted into cells L3:L17523, and then sorted from minimum to maximum. The 2.5% limit is then the one in row 0.022783*n +2, and the 97.5% limit is in row 0.977217*n +2. The 95/95 peak to trough distance is then the 97.5% value minus the 2.5% value.

A similar section, covering the limiting Period 5, is as follows:

	AE	AF	AG	AH
2	Period 5 (unsort/sort)			
3	99.62128	97.85396	2.5%	2.28%
4	99.62128	97.85396	438	399
5	99.62128	97.85396	98.14851	98.10643
6	99.57921	97.89604		
7	99.62128	97.89604	97.50%	97.72%
8	99.57921	97.89604	16991	17030
9	99.62128	97.89604	99.91583	99.91583
10	99.62128	97.89604	1.767324	1.809396

Here the good data from 1/1/2013 to 1/1/2015 are in cells AE3:AE17523. They are copied into column AF, sorted, and fill up cells AF3:AF17427 (fewer than in column AE because of the blanks associated with bad data). The data in column AG is a check; the 2.5% row is row 438, calculated as =ROUND(AG3*17425,0)+2. The 2.5% value is =AF438 and is 98.14851. Similarly, the 97.5% row is row

16991, and that value is 99.91583. The difference is 1.767324, which would be our 95% range if the data set included all possible data.

The actual results are in column AH, where we look at the range from the 2.28% row to the 97.72% row, and find a total difference of 1.81%. This is our 95/95 value for greatest peak to trough difference of any of our examined periods.

Aside: note that there was no numeric difference between the 97.5% row 16991 and the 97.7% row 17030. That is not unusual. The data is reported to 5 digits after the decimal, but the actual values are limited to a set of values. Hence in this case, rows 16835 through 17353 were all 99.91583.

Conclusion

The final results are shown below.

Period	Two year periods (1/1 to 1/1)	2.5% limit	97.5% limit	95/95 Range
1	2009-2011	98.61	100.13	1.51
2	2010-2012	98.53	99.87	1.35
3	2011-2013	98.49	99.96	1.47
4	2012-2014	98.19	99.92	1.73
5	2013-2015	98.11	99.92	1.81
6	2014-2016	98.02	99.79	1.77

The limiting 95/95 range is +/-1.81%. This value can be used as a non-random drift for all Columbia GS B-PPC points.

The quantified analysis and data are contained in the Excel file for the B-PPC eDNA Excel drift data for 95-95 analysis.xlsx. This file can be found in the EDMS folder of CGS AR 344042. The excel quantified analysis and data was developed for the 95/95 analysis from the PPC excel spreadsheet attached to the CGS transmittal document for B-PPC drift data as file B-PPC eDNA 1-1-09 to 5-16-16 per AR 344042-69 and also included in the EDMS folder of AR 344042.

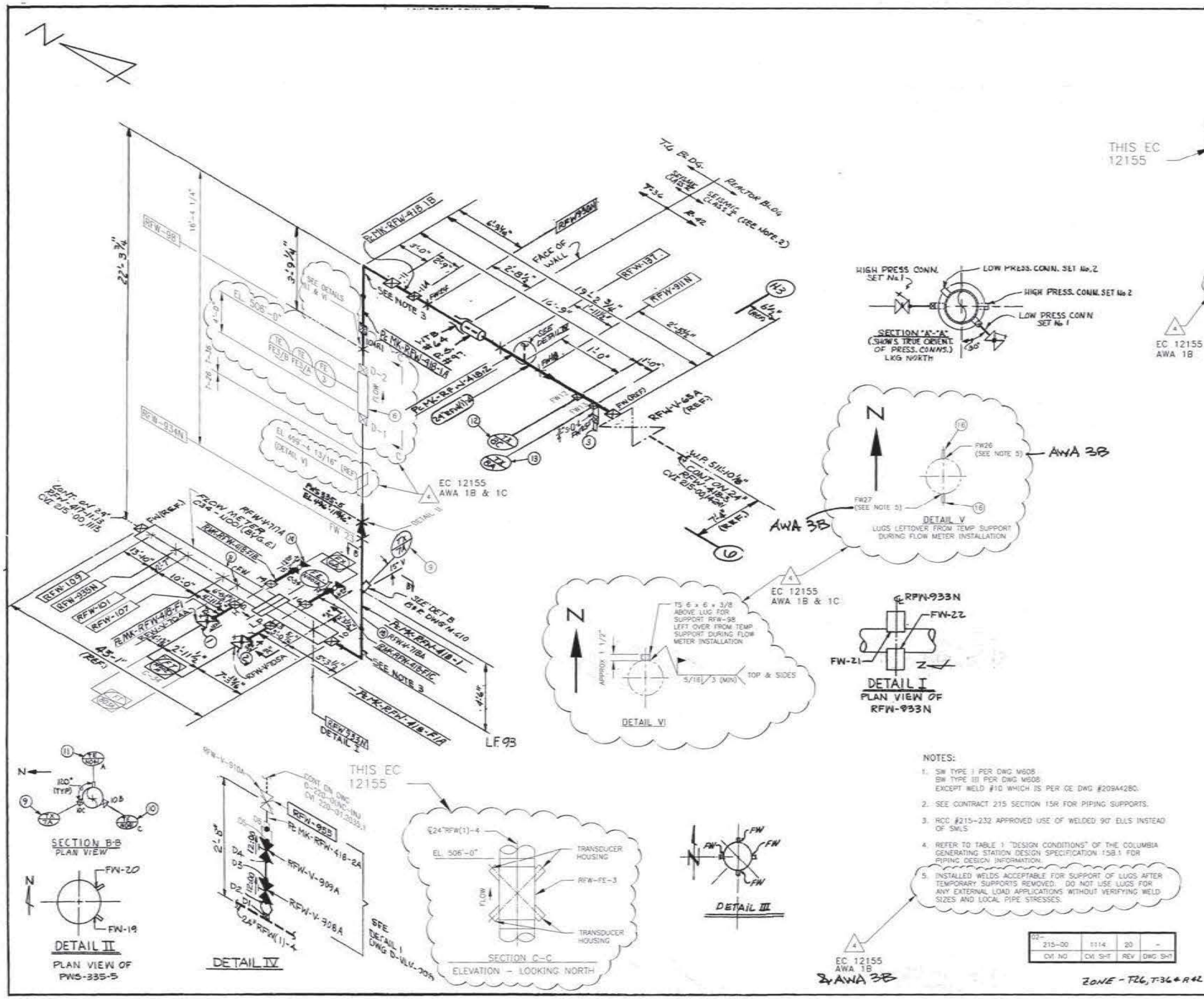
License Amendment Request to Revise Operating License and Technical Specifications for Measurement Uncertainty Recapture (MUR) Power Uprate

Enclosure 14

LEFM Flowmeter Installation Drawings

RFW-419-1.2

RFW-418-1.2



NOTES:

1. SW TYPE I PER DWG M608
   BW TYPE III PER DWG M608
   EXCEPT WELD #10 WHICH IS PER GE DWG #209A4280.
2. SEE CONTRACT 215 SECTION 15R FOR PIPING SUPPORTS.
3. RCC #215-232 APPROVED USE OF WELDED 90° ELLS INSTEAD OF SMLS
4. REFER TO TABLE 1 "DESIGN CONDITIONS" OF THE COLUMBIA GENERATING STATION DESIGN SPECIFICATION 158.1 FOR PIPING DESIGN INFORMATION.
5. INSTALLED WELDS ACCEPTABLE FOR SUPPORT OF LUGS AFTER TEMPORARY SUPPORTS REMOVED. DO NOT USE LUGS FOR ANY EXTERNAL LOAD APPLICATIONS WITHOUT VERIFYING WELD SIZES AND LOCAL PIPE STRESSES.

02-215-00	1114	20	-
CVI NO	CVI SHT	REV	DWG SHT

LIST OF MATERIAL

ITEM	QTY	SIZE	DESCRIPTION	HEAT NO
1	1	3/4"	GLOBE VALVE, 1500# C.S., S.W. RFW-V-704A	
2	1	3/4"	GLOBE VALVE, 1500# C.S., S.W. RFW-V-705A	
3	1	3/4"	PLUG, 6000# S.W. C.S. A-181 GR.II	
6	1	24"	CAMERON ULTRASONIC FLOW METER,	
			CARBON STEEL, BW, RFW-FE-3	
			— LOOSE MAT'L —	
8	1	24"	FLOW METER SECTION C34-N001	
			(BY G.E.) RFW-FE-N001A	
9	1	–	TEMP WELL FOR RFW-TX-7A	
10	1	–	TEMP WELL FOR RFW-TE-N041C	
11	1	–	TEMP WELL FOR RFW-TE-N041A	
12	1	–	TEMP TEST POINT FOR RFW TX-8C	
13	1	–	TEMP TEST POINT FOR RFW-TX-8A	
14	1	3/4"	GLOBE VALVE, 1500#, S.W, C.S, RFW-V-717A	
15	1	3/4"	GLOBE VALVE, 1500#, S.W C.S. RFW-V-718A.	
16	2		PL 1/2" x 3" x 3", A36 (LUGS LEFT OVER FROM TEMP SUPPORT	
			FROM INSTALLATION OF FLOW METER)	

THIS EC 12155

EC 12155 AWA 1B

DESIGN DOCUMENT CHANGE REVISION

PREPARED BY/DATE: DR JACOBSEN 6-15-15 — EC REV: 4
EX. ENG. CHKD BY/DATE: N/R
PHYS. DESIGN CHKD BY/DATE: [signature] 6/16/15
26145R R0

DESIGN DOCUMENT CHANGE

EC Number: 12155	Page No: 441
Doc Number: 215-00,1114	Rev: 20
Zone: N/A	
Prepared by: R. JEFFREY	Date: 12-12-13
*Technical Eng. Checked by: N/R	Date:
Physical Design Checked by: D.E. BANKS	Date: 3/11/14

* Required for Functional Design Documents (e.g. Flow Diag., One Line Diag's and Elementary Diag's, etc.) Ref: DES-3-5. If not required, indicate N/R.

26145m R2

REV	DATE	DESCRIPTION	DWN	CHK	APVD
20	6-30-2011	REV PER 7889-306 AND 10235 SCANNED DRAWING	SRP	DRJ	CBS

REVISIONS

QUALITY CLASS	II	SEISMIC CLASS	II, I (NOTE 2)
CODE GROUP	D	CLEANLINESS CLASS	B
DESIGN TEMP	450°F	DESIGN PRESS	1950 PSIG
MATERIAL SPEC	15B.1-344	NOTE	
WELD END PREP	SEE NOTE 1		
WELD PROCEDURE			
ROOT PASS			
COVER PASS			
HANGER GROUP	B&R 150, M-200 SHT 335		
PAINT			
INSULATION			
FPC		CODE	-
TYPE OF INSPECTION REQUIRED			

SYMBOLS

RFW 72.1

S/U

NON-DESTRUCTIVE EXAMINATION

MT	PT	RT	UT		

INSPECTOR IDENTIFICATION

INIT	FULL NAME	CO

WPPSS HANFORD NO 2
BOVEE & CRAIL CONSTRUCTION CO
GENERAL ENERGY RESOURCES INC
JOINT VENTURE

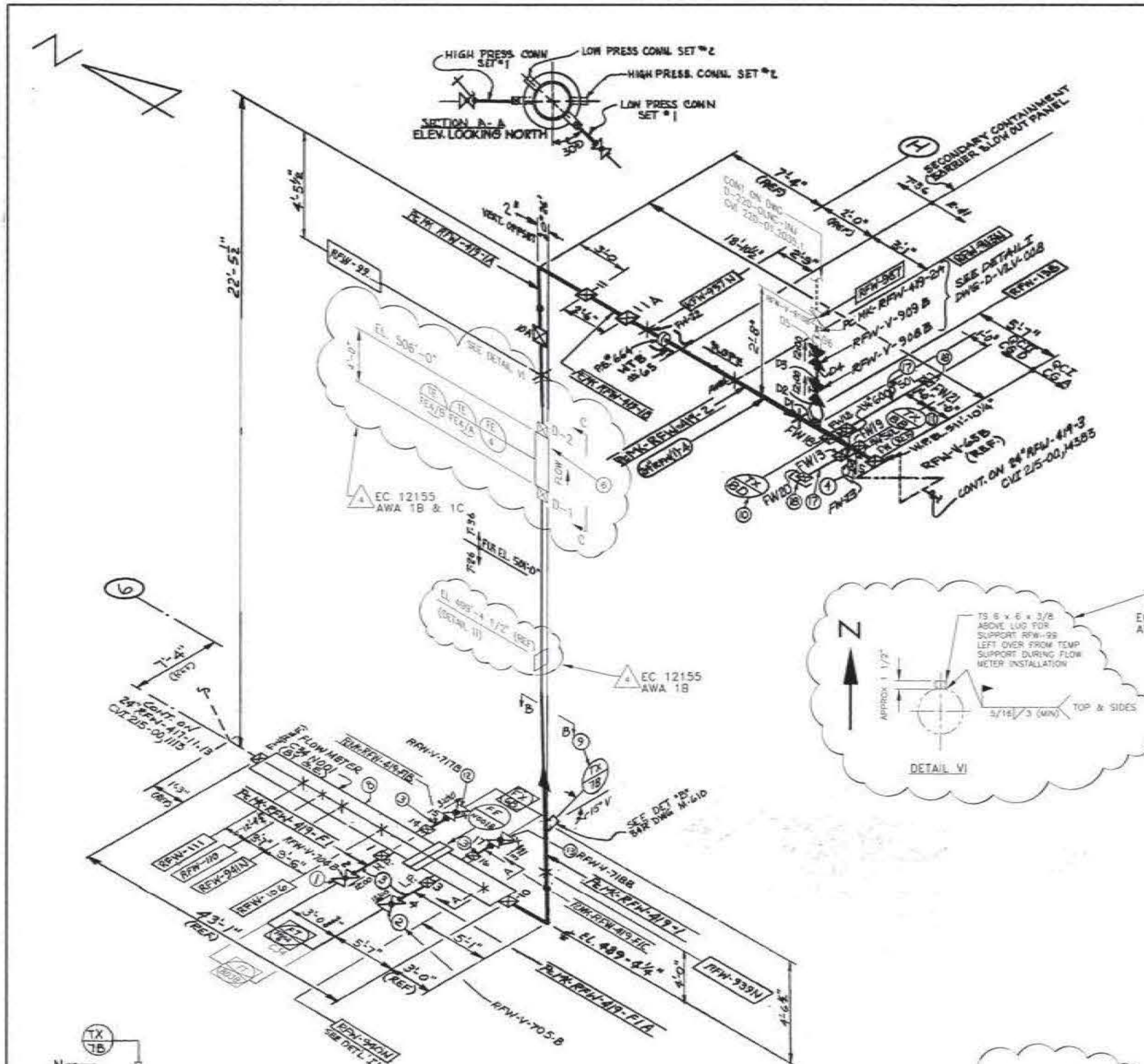
FLOW DIA	M-504-2 M-529
REF DWG	M-642 M-644 M-713
VENDOR DWG	M-646

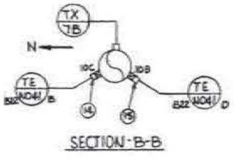
TITLE FROM FLOW METER TO REACTOR VESSEL (LINE 'A') — C

Dwn by	Date	Chk by	Date
HELM	12-2-74	SRW	7-15-75
Appv'd by	Date 7-17-75	Rel for Fab	Date

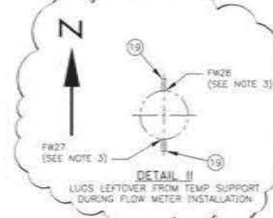
DWG. NO RFW-418-1.2 — REV 20

FORM NF 3





LP. 98
ZONE: T-26, T-36, R-41



EC 12155
AWA 1B

NOTES:

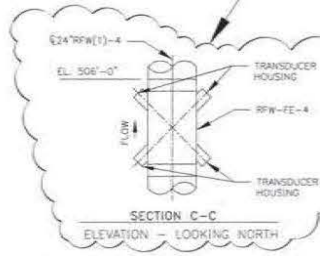
1. SW TYPE I PER DWG M608.
B.W. TYPE III PER DWG M608
EXCEPT WELD NO. 10 WHICH IS PER G.E. DWG # 209A4260.

2. REFER TO TABLE 1 "DESIGN CONDITIONS" OF THE COLUMBIA GENERATING STATION DESIGN SPECIFICATION 15B.1 FOR PIPING DESIGN INFORMATION.

EC 12155
AWA 1B

3. INSTALLED WELDS ACCEPTABLE FOR SUPPORT OF LUGS AFTER TEMPORARY SUPPORTS REMOVED. DO NOT USE LUGS FOR ANY EXTERNAL LOAD APPLICATIONS WITHOUT VERIFYING WELD SIZES AND LOCAL PIPE STRESSES.

THIS EC
12155



HGR INFORMATION	
RFW-111	SPRING
RFW-110	SPRING
RFW-941N	RIGID STRUT
RFW-106	SPRING
RFW 940N	SNUBBER
RFW-939N	RIGID STRUT
RFW-99	SPRING
RFW-937N	RIGID STRUT
RFW-913N	SNUBBER
RFW-138	SPRING
RFW-957	LUG (2)

THIS EC
12155

LIST OF MATERIAL

ITEM	QT'Y	SIZE	DESCRIPTION	HEAT NO
1	1	3/4"	GLOBE VALVE, 1500# C.S. S.W. (RFW-V-704B)	
2	1	3/4"	GLOBE VALVE, 1500# C.S. S.W. (RFW-V-705B)	
3	4	3/4"	× 6" LG. SMLS. PIPE, SCHED. 160, C.S.	
			A 106 GR. B, PBE	
4	1	3/4"	PLUG 6000# F.S. SW; A-105	
6	1	24"	CAMERON ULTRASONIC FLOW METER,	
			CARBON STEEL, BW, RFW-FE-4	
			— LOOSE MAT'L —	
8	1	24"	FLOW METER SECTION C34-N001	
			(BY G.E.) RFW-FE-N001 B	
9	1	-	TEMP WELL FOR RFW-TX-7B	
10	1	-	TEMP WELL FOR RFW-TX-8D	
11	1	-	TEMP WELL FOR RFW-TX-8B	
12	1	3/4"	GLOBE VALVE, 1500#, S.W. C.S. RFW-V-717B.	
13	1	3/4"	GLOBE VALVE, 1500#, S.W. C.S., RFW-V-718B.	
14	1	-	TEMP. WELL FOR RFW-TE-N041-B.	
15	1	-	TEMP. WELL FOR RFW-TE-N041-D.	
16	1	24"	x 2'-6" LG. PIPE, SCH. 120, SMLS, C.S., A106 GR.B	
17	2	1 1/4"	x 6" LG. SMLS. PIPE SCH 160, CS, A106 GR. B PBE	
18	2	1 1/4"	CAP 6000# SW CS A-105	
19	2		℄ 1/2" x 3" x 3", A36 (LUGS LEFT OVER FROM TEMP SUPPORT	
			FROM INSTALLATION OF FLOW METER)	

EC 12155
AWA 1B

DESIGN DOCUMENT CHANGE REVISION

PREPARED BY/DATE: DR JACOBSEN 6-15-15 EC REV 4
TECH. ENG. CHKD BY/DATE: N/R
PHYS. DESIGN CHKD BY/DATE: [signature] 6/16/15
26145R R0

DESIGN DOCUMENT CHANGE	
EC Number: 12155	Page No: 443
Doc Number: 215-00,1115	Rev: 14
Zone: N/A	
Prepared by: R. JEFFREY	Date: 12-12-13
*Technical Eng. Checked by: N/R	Date:
Physical Design Checked by: D.E. BANKS	Date: 3-11-14
* Required for Functional Design Documents (e.g. Flow Diag., One Line Diag's and Elementary Diag's, etc.) Ref. DES-5-5. If not required, indicate N/R	

26145H R2

Q2- 215-00	1115	14	—
CVI NO	CVI SHT	REV	DWG SHT

REV	DATE	DESCRIPTION	DWN	CHK	APVD
14	6-30-2011	REV PER 7889-315 AND 10235	SRP	DRJ	CBS

REVISIONS

QUALITY CLASS	II	SEISMIC CLASS	II
CODE GROUP	D	CLEANLINESS CLASS	B
DESIGN TEMP	450°F	DESIGN PRESS	1950 PSIG
MATERIAL SPEC	15B.1-344	NOTE	
WELD END PREP	SEE NOTE 1		
WELD PROCEDURE			
ROOT PASS			
COVER PASS			
HANGER GROUP	B&R 150 M-200 SHT 335		
PAINT			
INSULATION			
FPC		CODE	
TYPE OF INSPECTION REQUIRED			

SYMBOLS

BBE BEVEL BOTH ENDS
BOE BEVEL ONE END
BOP BOTTOM OF PIPE ELEV
FW FIELD WELD
H HORIZONTAL
PBE PLAIN BOTH ENDS
POE PLAIN ONE END
MQL MATERIAL QUALITY REQUIREMENTS
SOE SHAPE ONE END
SW SHOP WELD
TBE THREAD BOTH ENDS
TL TRUE LENGTH
TOE THREAD ONE END
TOP TOP OF PIPE ELEV
V VERTICAL
WP WORK POINT
FIELD WELD
MANUAL SHOP WELD
VERIFY DIMENSION IN FIELD

SJU-72.1

NON-DESTRUCTIVE EXAMINATION

MT	RT	RT	UT		

WPPSS HANFORD NO 2
BOVEE & CRAIL CONSTRUCTION CO
GENERAL ENERGY RESOURCES INC
JOINT VENTURE

FLOW DIA	M504-2 M-529
REF DWG	M-642 M-644 M-523
VENDOR DWG	M-646

TITLE REACTOR FEEDWATER FROM FLOWMETER TO REACTOR VESSEL (LINE B) C

INSPECTOR IDENTIFICATION

INIT	FULL NAME	COD

Dwn by HEKM	Date 11-26-74	Chk by SRW	Date 3-15-75
Appd by	Date 7-19-75	Rel for Fab	Date

DWG NO RFW-419-1.2 REV 14