J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP | 7800 Rochester Hwy
Seneca, SC 29672

o: 864.873.3478
f: 864.873.5791
Ed.Burchfield@duke-energy.com

RA-19-0144 10 CFR 50.90

February 19, 2020

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

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2 and 3
Renewed Facility Operating Licenses Numbers DPR-38, DPR-47, and DPR-55
Docket Numbers 50-269, 50-270, and 50-287

License Amendment Request for Measurement Uncertainty Recapture Power Uprate

References:

1. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation, dated September 20, 2011, License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02 (ML11269A127)
2. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation dated November 21, 2011, Supplement to License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02, Supplement 1 (ML11326A296)
3. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation dated March 16, 2012, Supplement to License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02, Supplement 2 (ML12109A345)
4. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation dated April 4, 2012, Supplement to License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02, Supplement 3

5. Letter from the U. S. Nuclear Regulatory Commission to Preston Gillespie, Site Vice President, Oconee Nuclear Station, Duke Energy Carolinas dated August 31, 2012, Request for Additional Information and Suspension of Review of License Amendment Request for a Measurement Uncertainty Recapture Power Uprate (ML12234A558)
6. NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002 (ML013530183)

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend the Technical Specifications (TS) of Renewed Facility Operating License Nos. DPR-38, DPR-47 and DPR-55 to support a measurement uncertainty recapture (MUR) power uprate. This MUR power uprate License Amendment Request (LAR) would increase each unit's authorized core power level from 2568 megawatts thermal (MWt) to 2610 MWt; an increase of 42 MWt and approximately 1.64% of Rated Thermal Power (RTP). The U.S. Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10 CFR 50, Appendix K that provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the emergency core cooling system (ECCS) evaluation, or applying an appropriately justified reduced margin for ECCS evaluation. Based on the use of the Cameron (a.k.a. Caldon) instrumentation to determine core power level with a power measurement uncertainty of approximately 0.34 percent, Duke Energy proposes to reduce the licensed power uncertainty within the requirements of 10 CFR 50, Appendix K, resulting in an approximately 1.64% increase in megawatts thermal.

Specifically, this LAR requests NRC approval for certain Oconee Nuclear Station (ONS) Technical Specification (TS) changes necessary to support operation at the uprated power level.

Enclosure 1 provides an evaluation of the proposed changes, the determination that the proposed amendment contains No Significant Hazards Consideration and the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement pursuant to 10 CFR 51.22(c)(9). Enclosure 2 provides a technical review of the proposed power uprate formatted to align with RIS 2002-03 (Reference 6) guidance sections.

This LAR supersedes the previous MUR LAR dated September 20, 2011 (Reference 1) and its associated documentation as well as the associated responses (References 2, 3, and 4) to NRC Requests for Additional Information (RAI). The NRC suspended review of the previous MUR LAR by letter dated August 31, 2012 (Reference 5). Duke Energy has incorporated the RAI responses from the previous MUR LAR submittal, as appropriate, and addressed lessons learned from NRC review of recent MUR LARs, as applicable. This LAR also addresses RAI questions that were transmitted with the suspension letter.

Attachment 1 provides a list of regulatory commitments being made as a result of this LAR. Attachment 2 contains a marked up version of the affected Facility Operating License (FOL) and TS pages. Attachment 3 contains retyped FOL and TS pages. Attachments 4 and 5 contain marked up and retyped TS Bases changes, respectively, for information only.

Attachments 6 and 8 contain non-proprietary and proprietary Cameron reports, respectively. As Attachment 8 contains information proprietary to Cameron, it is supported by affidavits signed by Cameron, the owner of the information. The Cameron affidavits, provided in Attachment 7, set forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390. Accordingly, it is requested that the information that is proprietary to Cameron be withheld from public disclosure in accordance with 10 CFR 2.390.

Duke Energy requests approval of this amendment request by February 19, 2021. Once approved, the amendment will be implemented within 120 days. The first MUR power uprate will occur following the Unit 2 Fall 2021 refueling outage.

In accordance with Duke Energy administrative procedures and the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the On-site Review Committee. A copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Please refer any questions regarding this submittal to Art Zaremba, Director - Fleet Licensing, at (980) 373-2062.

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

Executed on February 19, 2020.

Very truly yours,

J Ed Burchfield Jr

J. Ed Burchfield, Jr.
Vice President
Oconee Nuclear Station

Enclosure 1 Evaluation of Proposed Changes
Enclosure 2 RIS 2002-03 Requested Information

Attachment 1	Regulatory Commitments
Attachment 2	Facility Operating License and Technical Specification Markups
Attachment 3	Retyped Facility Operating License and Technical Specifications
Attachment 4	Marked Up Technical Specifications Bases (for information only)
Attachment 5	Retyped Technical Specification Bases (for information only)
Attachment 6	Non-Proprietary Cameron Reports
Attachment 7	Cameron Affidavits
Attachment 8	Proprietary Cameron Reports

cc w/enclosures and attachments:

Ms. Laura A. Dudes, Administrator, Region II
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257

Ms. Audrey Klett, Project Manager
(by electronic mail only)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O-8G9A
11555 Rockville Pike
Rockville, Maryland 20852

Mr. Jared Nadel
NRC Senior Resident Inspector
Oconee Nuclear Station

cc w/enclosures and Attachments 1 through 6:

Ms. Anuradha Nair-Gimmi,
(by electronic mail only: naira@dhec.sc.gov)
Bureau Environmental Health Services
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

ENCLOSURE 1 EVALUATION OF PROPOSED CHANGES

Subject: Proposed License Amendment Request to support a measurement uncertainty recapture (MUR) power uprate.

1 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend the Technical Specifications (TS) and the Renewed Facility Operating Licenses (OLs) Nos. DPR-38, DPR-47 and DPR-55 for Oconee Nuclear Station (ONS) Units 1, 2, and 3. This License Amendment Request (LAR) proposes to increase each unit's authorized core power level from 2568 megawatts thermal (MWt) to 2610 MWt; an increase of approximately 1.64% Rated Thermal Power.

Selected Licensee Commitments (SLCs) and the UFSAR will be changed as required to support the power uprate following implementation of the MUR power uprate.

1.1 Background

ONS Units 1, 2, and 3 are presently licensed for a core power rating of 2568 MWt. Duke Energy is seeking to increase the licensed core power to 2610 MWt an increase of approximately1.64%, based on the use of more accurate feedwater flow measurement instrumentation.

The 1.64% core power uprate for ONS Units 1, 2, and 3 (hereby referred to as the Measurement Uncertainty Recapture (MUR) Power Uprate) is based on recapturing measurement uncertainty currently included in the analytical margin originally required for emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR) 10 CFR 50, Appendix K (Emergency Core Cooling System Evaluation Models, ECCS).

The U.S. Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10 CFR 50, Appendix K that provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying an appropriately justified reduced margin for ECCS evaluation.

Based on the use of the Cameron (a.k.a. Caldon) instrumentation to determine core power level with a power measurement uncertainty of approximately 0.34 percent (the allowable increase of 1.66% Rated Thermal Power truncated to the lower whole megawatt is a 1.64% uprate), Duke Energy proposes to reduce the licensed power uncertainty within the requirements of 10 CFR 50, Appendix K, resulting in an approximately 1.64% increase in megawatts thermal.

The impact of the MUR Power Uprate has been evaluated on the plant systems, structures, components, safety analyses, and off-site interfaces. Enclosures 1 and 2 to this License Amendment Request summarize these evaluations, analyses, and conclusions.

In conjunction with the installation of the Cameron CheckPlus leading edge flow meter (LEFM), two additional design changes are required to be implemented to support operation at the uprated power level:

- Control Rod Drive Mechanism (CRDM) re-power – this modification is credited in the updated HELB analysis discussed in Section II.1.D.iii.29.
- Safe Shutdown Facility (SSF) Letdown Line – this modification is credited in the updated Feedwater Line Break (FWLB) and High Energy Line Break (HELB) analysis discussed in Section II.D.iii.27 and 29.

The modifications are being performed under 10 CFR 50.59 and their approval is not part of this LAR. These modifications will be completed prior to the MUR power uprate on each ONS unit and are identified as Regulatory Commitments in Attachment 1.

2 DETAILED DESCRIPTION

2.1 Description of Proposed Changes

To accommodate a rated thermal power level of 2610 megawatts thermal for ONS Units 1, 2, and 3, Duke Energy proposes to modify the Facility Operating License and Technical Specifications. The proposed changes are described below:

Facility Operating Licenses Page 3 – Maximum Power Level

For the ONS Units 1, 2, and 3 facility operating licenses, the steady state licensed power level will change from 2568 MWt to 2610 MWt, corresponding to the new RTP in the Technical Specifications.

TS 1.1, Definition of Rated Thermal Power

RATED THERMAL POWER will change from 2568 MWt to 2610 MWt.

Since MUR Uprate will be implemented on a staggered basis for each unit, this change is accomplished using a temporary footnote which indicates, following implementation of MUR on the respective unit, the value of RATED THERMAL POWER shall be 2610 MWt.

TS Table 3.3.1-1

Reduce the Nuclear Overpower High Setpoint for three pumps operating from ≤ 80.5% RTP to ≤ 79.3% RTP. Add a footnote “f” to say: “If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in Selected Licensee Commitment 16.7.18, Leading Edge Flow Meter (LEFM).” Added footnote “g” to say: “Following implementation of MUR on the respective Unit, the value of RTP shall be 79.3%.

Since MUR Uprate will be implemented on a staggered basis for each unit, the setpoint change is accomplished using temporary footnote (g) which indicates, following implementation of MUR on the respective unit, the value of RTP shall be 79.3%.

TS Figures 3.4.3-1 through 3.4.3-9

The Applicability for the RCS Heatup and Cooldown limit curves is revised to 44.6 Effective Full Power Years (EFPY) for Unit 1, to 45.3 EFPY for Unit 2, and to 43.8 EFPY for Unit 3 based on updated reactor vessel (RV) material evaluations discussed in Section IV.1.

TS 3.4.4, Reactor Coolant System (RCS) Loops – MODES 1 and 2

For three reactor coolant pumps (RCPs) operating reduce the thermal power limit from ≤ 75% RTP to ≤ 73.8% RTP.

Since MUR Uprate will be implemented on a staggered basis for each unit, this change is accomplished using a temporary footnote which indicates, following implementation of MUR on the respective unit, the value of RATED THERMAL POWER shall be ≤ 73.8% RTP.

Selected Licensee Commitments (SLCs)

As discussed in Enclosure 2, a Selected Licensee Commitment (SLC) is being added to support implementing this LAR. The new SLC adds functionality requirements for the leading edge flow meters and appropriate Required Actions and Completion Times when an LEFM is not functional. This SLC includes a 72-hour Completion Time for restoring a nonfunctional LEFM which is consistent with SLCs for the Cameron LEFM CheckPlus systems at McGuire Units 1 and 2 and Catawba Unit 1.

Attachment 2 contains a marked-up version of the affected Facility Operating License (FOL) and TS pages. Attachment 3 contains retyped FOL and TS pages. Attachments 4 and 5 contain marked up and retyped TS Bases changes, respectively.

2.2 Reason for the Proposed Changes

The proposed changes will allow Duke Energy to increase the Rated Thermal Power (RTP) from 2568 MWt to 2610 MWt.

3 TECHNICAL EVALUATION

Oconee Units 1, 2, and 3 are presently licensed for a Rated Thermal Power (RTP) of 2568 MWt. A more accurate feedwater flow measurement supports a 1.64% increase to 2610 MWt. The technical evaluation for this MUR power uprate, summarized in Enclosure 2, addresses the following aspects:

- the feedwater flow measurement technique and power measurement uncertainty,
- accidents and transients that remain bounded at the proposed uprated power level,
- accidents and transients that are not bounded at the proposed uprated power level,
- mechanical/structural/material component integrity and design,
- electrical equipment design,
- system design,
- operating, emergency, and abnormal procedures including associated operator actions,
- environmental impact, and
- changes to the Technical Specifications including protective system setpoints.

The evaluation conclusions are summarized in Enclosure 2, which is formatted to align with the NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 0) guidance sections.

In addition, Duke Energy evaluated the potential impact of License Amendment Requests (LARs) that were recently approved, submitted and awaiting NRC approval, or in an in-process status. None adversely impact the MUR. The MUR was determined to not adversely impact LARs that have been submitted and are awaiting NRC approval.

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

RIS 2002-03 provides generic guidance for evaluating an MUR power uprate. Enclosure 2 to this request for a license amendment provides the ONS specific evaluation of each step outlined in RIS 2002-03, Attachment 1, and provides a description of the methodology used by ONS to complete the evaluation. Based on Enclosure 2, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation at the uprated power level, (2) operation at the uprated power level will comply 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.

4.2 Precedent

This request is similar in format and content to the following four submittals:

1. PSEG Nuclear submittal for measurement uncertainty recapture power uprate of the Hope Creek Generating Station, dated July 7, 2017 (ML17188A259), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment, dated April 24, 2018 (ML18096A542).
2. Exelon Generation submittal for measurement uncertainty recapture power uprate of the Peach Bottom Atomic Power Station dated February 17, 2017 (ML17048A444), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment, dated November 15, 2017 (ML17286A013).

3. Energy Northwest submittal for measurement uncertainty recapture power uprate of the Columbia Generating Station dated June 28, 2016 (ML16183A365), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated, May 11, 2017 (ML17095A117).
4. Duke Energy submittal for measurement uncertainty recapture power uprate of the Catawba Nuclear Station, dated June 23, 2014 (ML14176A109), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated April 29, 2016 (ML16081A333).

The BWR MURs listed above were reviewed for topics common to BWR and PWR designs including the Cameron LEFM, reactor vessel materials, electrical equipment design, and Flow Accelerated Corrosion.

4.3 Significant Hazards Consideration

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment to Oconee Nuclear Station (ONS) Units 1, 2, and 3 Facility Operating Licenses DPR-38, -47, and -55 by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

The requested change will affect certain Technical Specifications by increasing the rated thermal power level. Technical Specification (TS) changes are discussed in Section 3 above and detailed markups are included in Attachment 2 to this License Amendment Request.

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

   Response: No.

   The proposed amendment changes the rated thermal power from 2568 megawatts thermal (MWt) to 2610 MWt; an increase of approximately 1.64% Rated Thermal Power. Duke Energy's evaluations have shown that all structures, systems and components (SSCs) are capable of performing their design function at the uprated power of 2610 MWt. A review of station accident analyses found that all acceptance criteria are still met at the uprated power of 2610 MWt.

   The radiological consequences of operation at the uprated power conditions have been assessed. The proposed power uprate does not affect release paths, frequency of release, or the analyzed reactor core fission product inventory for any accidents previously evaluated in the Updated Final Safety Analysis Report. Analyses performed to assess the effects of mass and energy releases remain valid. All acceptance criteria for radiological consequences continue to be met at the uprated power level.

   The proposed change does not involve any change to the design or functional requirements of the safety and support systems. That is, the increased power level neither degrades the performance of, nor increases the challenges to any safety systems assumed to function in the plant safety analysis.

   While power level is an input to accident analyses, it is not an initiator of accidents. The proposed change does not affect any accident precursors and does not introduce any accident initiators. The proposed change does not impact the usefulness of the Surveillance Requirements (SRs) in evaluating the operability of required systems and components.

   In addition, evaluation of the proposed TS changes demonstrates that the ability of equipment and systems required to prevent or mitigate the radiological consequences of an accident is not significantly affected. Since the impact on the systems is minimal, it is concluded that the overall impact on the plant safety analysis is negligible.

   Therefore, the proposed TS change does not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment 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 single failures are introduced as a result of the proposed change. The installation of the Cameron LEFM CheckPlus System has been analyzed and failures of the system will have no adverse effect on any safety-related system or any systems, structures, and components (SSCs) required for transient mitigation. SSCs previously required for the mitigation of a transient continue to be capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety-related system or component and does not change the performance or integrity of any safety-related system.

   The proposed change does 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. Operation at the uprated power level does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed change.

   Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   Although the proposed amendment increases the ONS Units 1, 2, and 3 operating power level, the units retain their margin of safety because it is only increasing power by the amount equal to the reduction in uncertainty in the heat balance calculation. The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, reactor coolant system pressure boundary, and containment barriers. Analyses demonstrate that the current design basis continues to be met after the measurement uncertainty recapture (MUR) power uprate. Components associated with the reactor coolant system pressure boundary structural integrity, including pressure-temperature limits, vessel fluence, and pressurized thermal shock are bounded by the current analyses. Systems will continue to operate within their design parameters and remain capable of performing their intended safety functions.

   The current ONS safety analyses including the design basis radiological accident dose calculations, bound the power uprate.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

4.4 Conclusions

Duke Energy has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92 in Section 5.1 of this Enclosure.

The regulatory requirements and guidance applicable to this LAR are identified in Section 4.3 above.

Duke Energy identified several LARs, as indicated in Section 4.2 above, requesting measurement uncertainty recapture power uprates. These LARs used the applicable regulatory requirements of Section 4.1 above to provide a basis for NRC review and approval. Duke Energy used these LARs to the extent practical and applicable for developing this LAR.

5 ENVIRONMENTAL CONSIDERATION

Duke Energy has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21 (See Section VII.5 of Enclosure 2). Duke Energy has determined that this license amendment request meets the criteria for a categorical exclusion as set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that the amendment meets the following specific criteria:

1. The amendment involves no significant hazard consideration as demonstrated in Section 4.3 above.
2. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite. The principal barriers to the release of radioactive materials are not modified or affected by this change and no significant increases in the amounts of any effluent that could be released offsite will occur as a result of this change.
3. There is no significant increase in individual or cumulative occupational radiation exposure. Because the principal barriers to the release of radioactive materials are not modified or affected by this change, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment pursuant to 10 CFR 51.22(b).

6 REFERENCES FOR ENCLOSURE 1

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

ENCLOSURE 2 RIS 2002-03 REQUESTED INFORMATION

This enclosure provides information in response to each item of RIS 2002-03, Attachment 1.

TABLE OF CONTENTS for ENCLOSURE 2:

Page

ACRONYMS

AB	Auxiliary Building
AC	Alternating current
ADV	Atmospheric Dump Valve
ALARA	As Low As Reasonably Achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
AOR	Analysis of record
AOT	Allowable Outage Time
AOV	Air Operated Valve
ART	Adjusted Reference Temperature
ARTS	Anticipatory Reactor Trip System
AS	Auxiliary Steam
ASME	American Society of Mechanical Engineers
ASW	Auxiliary Service Water
ATWS	Anticipated Transient Without Scram
AV	Allowable Value (in Technical Specifications)
BOP	Balance-of-plant
BS	Reactor Building Spray
BWC	Babcock & Wilcox Canada
BWST	Borated water storage tank
CA	Chemical Addition
CC	Component Cooling
CCVT	Coupling Capacitor Voltage Transformer
CCW	Condenser Circulating Water
CF	Core Flood
CFM	Centerline Fuel Melt
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
COGP	Combustible Gas Analyzer Program
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CRVS	Control Room Ventilation System
CS	Coolant Storage
DBA	Design basis accident
DBE	Design basis event
DC	Direct current
DHR	Decay Heat Removal
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DSLB	Double steam line break
DSS	Diverse Scram System
EAB	Exclusion area boundary
ECCW	Emergency Core Cooling Water
EFPY	Effective full-power years
EFW	Emergency Feedwater
EMA	Equivalent margin analysis
EOC	End of Cycle
EQ	Environmental Qualification
ES	Engineered Safeguards
ESF	Engineered Safety Feature
ETAP	Electrical Transient and Analysis Program

FAC	Flow-accelerated corrosion
FDW	Main feedwater
FIV	Flow induced vibration
GWD	Gaseous Waste Disposal
HELB	High energy line break
HEPA	High Efficiency Particulate
HPI	High Pressure Injection
HPSW	High Pressure Service Water
HZP	Hot Zero Power
ICS	Integrated Control System
INF	Inlet Nozzle Forging
IPB	Isolated Phase Bus
IS	Inner Shell
ISA	Instrument Society of America
ISI	Inservice Inspection
ISLH	Inservice leak and hydrostatic
IST	Inservice Testing
KHU	Keowee Hydro Unit
LAR	License Amendment Request
LBB	Leak-before-break
LBLOCA	Large Break Loss of Coolant Accident
LEFM	Leading Edge Flow Meter
LCO	Limiting Condition for Operation (Tech Specs)
LNB	Lower Nozzle Belt
LOCA	Loss of Coolant Accident
LOMFW	Loss of main feedwater
LOOP	Loss of Offsite Power
LPI	Low Pressure Injection
LPSW	Low Pressure Service Water
LTOP	Low temperature overpressure protection
LWD	Liquid Waste Disposal
M&E	Mass and energy
MFW	Main Feedwater
MFWLB	Main Feedwater Line Break
MIRVP	Master Integrated Reactor Vessel Surveillance Program
MOD	Motor Operated Disconnect
MOV	Motor Operated Valve
MS	Main Steam
MSLB	Main Steam Line Break
MSR	Moisture Separator Reheater
MSU	Main Step-up
MTC	Moderator Temperature Coefficient
MUR	Measurement Uncertainty Recapture
MVAR	1,000,000 VARs
MWe	Megawatts electric
MWt	Megawatts thermal
NFPA	National Fire Protection Association
NI	Nuclear Instrumentation
NPDES	National Pollution Discharge Elimination System
NPSH	Net positive suction head
NRC	(US) Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System

OAC	Operator Aid Computer
ODCM	Offsite Dose Calculation Manual
OEM	4kV Essential Auxiliary Power
ONS	Oconee Nuclear Station (Units 1, 2, and 3)
ONF	Outlet Nozzle Forging
OP	Operating Procedures
OTSG	Once Through Steam Generator
PAS	Post Accident Sampling
PCB	Power Circuit Breaker
PEPSE	Performance Evaluation of Power System Efficiencies
PSW	Protected Service Water
PTC	Performance Test Code (an ASME document)
PWROG	Pressurized Water Reactor Owners Group
RB	Reactor Building
RBC	Reactor Building Cooling
RCM	Reactor Coolant Makeup
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCW	Recirculating Cooling Water
REA	Rod Ejection Accident
RFS	Refueling System
RIS	Regulatory Issue Summary
ROTSG	Replacement Once Through Steam Generator
RP	Recommended Practice
RPS	Reactor Protection System
RTP	Rated Thermal Power (Licensed Power Level)
RT_{PTS}	Reference nil ductility transition temperature for pressurized thermal shock
RV	Reactor vessel
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SCD	Statistical Core Design
SE	Safety Evaluation
SER	Safety Evaluation Report
SF	Spent Fuel Cooling
SFP	Spent fuel pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SLC	Selected Licensee Commitments
SR	Surveillance Requirement
SRSS	Square Root of the Sum of the Squares
SSCs	Systems, Structures and Components
SSF	Standby Shutdown Facility
TBS	Turbine Bypass System
TCOA	Time Critical Operator Actions
TDEFWP	Turbine driven emergency feedwater pump
TID	Total integrated dose
TS	Technical Specification
UAT	Unit Auxiliary Transformer
UCC	Underclad Cracking
UFSAR	Updated Final Safety Analysis Report
US	Upper Shell
USE	Upper shelf energy

UT	Ultrasonic Testing
VAR	Volt-Ampere Reactive
WC	Chilled Water

I Feedwater flow measurement technique and power measurement uncertainty

I.1 A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. This description should include:

I.1.A Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

RESPONSE:

The feedwater flow measurement system at ONS Units 1, 2, and 3 is a Cameron (aka Caldon) CheckPlus Leading Edge Flow Meter (LEFM CheckPlus) with ultrasonic multi-path transit time flowmeter as described in the following topical reports:

Caldon, Inc. Engineering Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√ ™System,” Revision 0, March 1997 (Reference I.2)

Caldon® Ultrasonics Engineering Report: ER-157(P-A) Rev. 8 and Rev. 8 Errata, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System,” May 2008 (Reference I.3)

I.1.B A reference to the NRC’s approval of the proposed feedwater flow measurement technique

RESPONSE:

The Cameron Leading Edge Flow Meter Check instruments (Report ER-80P) were reviewed and approved by the NRC in the SER contained in letter 1 below. Subsequently, the Leading Edge Flow Meter Check Plus instruments (Report ER-157P-A, Revision 8, “Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System”) were reviewed and approved by the NRC in the SER in letter 2 below.

1. Letter from U.S. Nuclear Regulatory Commission to Terry C.L., TXU Electric, “Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report ER-80P, ‘Improving Thermal Power Accuracy and Plant Safety while Increasing Operating Power Level Using the LLEFM System,’ Comanche Peak Steam Electric Station, Units 1 and 2, (TACS Nos. MA2298 and MA2299),” March 8, 1999 (ADAMS Accession Number 9903190065, legacy library) (Reference I.4)

2. NRC letter from Thomas B. Blount, Deputy Director, NRC, to Mr. 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),” August 16, 2010 (ML102160663) (Reference I.5)

I.1.C **A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique**

RESPONSE:

The LEFM CheckPlus ultrasonic flow meter system will be installed and operated in accordance with the manufacturer's requirements as described in References I.2 and I.3. The system will be used for continuous calorimetric power determination by direct links with the ONS Units 1, 2, and 3 operator aid computers. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The LEFM CheckPlus ultrasonic flow meter system consists of an electronic cabinet located in the Turbine Building and two measurement section/spool pieces (each consisting of four electronic transmitters and four pressure transmitters), also located in the Turbine Building. The LEFMs are installed in horizontal runs of main feedwater piping upstream of the existing venturis. The ONS Unit 1 A and B LEFMs are 11.2 and 13.5 length/diameter (L/D) upstream of the existing venturis; the ONS Unit 2 A and B LEFMs are 12.9 and 15.0 L/D upstream of the existing venturis; and the ONS Unit 3 A and B LEFMs are 11.8 and 13.5 L/D upstream of the existing venturis. Caldon® Ultrasonics Engineering Report ER-755 (Reference I.20) presents and discusses calibration data obtained at Alden Test Laboratory to determine any effect of mounting an LEFM spool piece upstream of a flow measurement venturi. Tests were conducted both with a straight section of 20-inch diameter piping, and with a 20-inch diameter LEFM spool piece located upstream of the venturi. The venturi high pressure taps were located approximately 80 inches (L/D = 4) downstream of the discharge of the LEFM spool piece. This comparison showed that the LEFM spool piece had a negligible effect on the discharge coefficient of the venturi. No noticeable difference existed between the discharge coefficients for the model test versus the straight pipe test. The L/D values noted above for the location of each unit's LEFM of the existing venturi all exceed the L/D = 4 in the Caldon® Ultrasonics Engineering Report ER-755. It is concluded that the location of the LEFM spool pieces does not affect the downstream venturi.

The LEFMs meet or exceed the required 5 L/D downstream of elbows, laterals, or headers.

Testing of each of the ONS LEFM CheckPlus systems was performed at Alden Research Laboratories and the results are documented in Caldon® Engineering Report ER-855 (Reference I.6), which was included in the original LAR dated September 20, 2011 (Reference I.7) and is re-submitted in Enclosures 6 and 7 to this LAR. The piping arrangements (shown in Figures 1 and 2 of ER-855) reflect the installed locations of the LEFMs in ONS Units 1, 2, and 3. Each "A" loop LEFM is 7.2 L/D downstream from an upstream elbow (centerline to centerline) and each "B" LEFM is 5.0 L/D downstream from an upstream elbow (centerline to centerline). All elements of the lab measurements are traceable to National Institute for Standards and Technology standards.

I.1.D **The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique**

RESPONSE:

In approving Caldon, Inc. Engineering Report-80P, the NRC established four criteria to be addressed by each licensee. In approving Caldon® Ultrasonics Engineering Report ER-157P, Revision 8, the NRC

established five additional criteria to be addressed by each licensee. The following presents a discussion of each of the nine criteria relative to ONS Units 1, 2, and 3:

I.1.D.i **Criterion 1 from ER-80P - Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for unavailable LEFM instrumentation and the effect on thermal power measurements and plant operation.**

Maintenance and Calibration Procedures:

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation and maintenance at the uprated power level with the new LEFM CheckPlus system. Implementation will also include training of operating and maintenance personnel. A preventive maintenance program will be developed prior to implementing the LEFM CheckPlus system using Cameron's maintenance and troubleshooting manual and Duke Energy's established procedure program. Typical preventive maintenance activities include the following checks:

- General inspection of the terminal and cleanliness,
- Power Supply inspection of magnitude and noise,
- Central Processing Unit inspection,
- Acoustic Processor Unit Checks of the 5 MHz clock and LED status,
- Analog Input checks of the A/D converter,
- Alarm Relay checks,
- Watchdog Timer checks that ensures the software is running,
- Transducer Cable checks,
- Calibration checks of each of the Feedwater pressure transmitters.

The preventative maintenance program and continuous monitoring of the LEFM ensure that the LEFM operation remains bounded by the analysis and assumptions set forth by the LEFM vendor. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

Operation:

Details of ONS's proposed operation (including contingencies for LEFM unavailability) are discussed in response to Criterion 1 from ER-157P, Revision 8, below.

I.1.D.ii **Criterion 2 from ER-80P - For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed instrumentation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.**

ONS currently uses flow venturis to measure feedwater flow to support the secondary calorimetric power measurements. The LEFM CheckPlus measurement section/spool pieces have been installed as discussed in Section I.1.C. The electronic portion of the LEFM CheckPlus system was installed in January 2020 (Unit 2) and is scheduled to be installed in Units 1 and 3 later in 2020. After the LEFM CheckPlus system is installed and operational, 30 days of data will be collected comparing the LEFM

CheckPlus operating data to the venturi data to verify consistency between thermal power calculation based on LEFM data and other plant parameters.

I.1.D.iii **Criterion 3 from ER-80P - Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on 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 for comparison.**

The LEFM uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1 (Reference I.9) and Instrument Society of America (ISA) Recommended Practice (RP) ISA RP 67.04 (Reference I.10) and Alden Research Laboratory Inc. calibration tests. This methodology has been used for instrument uncertainty calculations for multiple MUR power uprates and has been indirectly approved by the NRC in the acceptance of those uprates and is consistent with the intent of plant setpoint methodology.

The feedwater flow and temperature uncertainties are combined with other plant measurement uncertainties (e.g., steam temperature, steam pressure, feedwater pressure) to calculate the overall heat balance uncertainty as described in Section I.1.E below. This LEFM uncertainty calculation method is consistent with the current heat balance uncertainty calculation that uses the feedwater flow venturis and RTDs.

I.1.D.iv **Criterion 4 from ER-80P – Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant specific installations), should provide additional justification for its use. This 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, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.**

This criterion does not apply to ONS, as the flow elements were tested and calibrated in a full-scale model of the ONS Units 1, 2, and 3 hydraulic geometry at the Alden Research Laboratory. A bounding calibration factor for the ONS Units 1, 2, and 3 spool pieces was established by these tests and is included in the Cameron engineering report for each unit (Caldon® Ultrasonics Engineering Reports ER-813 for ONS-1 (References I.13 and I.14), ER-824 for ONS-2 (References I.15 and I.16), and ER-825 for ONS-3 (References I.17 and I.18) are included in Attachments 6 and 8 to this LAR). A Caldon® Ultrasonics engineering report (ER-855 (References I.6 and I.21), included in Attachments 6 and 8 to this LAR), summarizes the testing and evaluates the test data. Caldon® Ultrasonics Engineering Report ER-972 (Reference I.9) contains a detailed cross reference of the sections in the Caldon topical reports (References I.2 and I.3) to the applicable sections in the plant-specific reports ER-813, ER-824, and ER-825. A bounding uncertainty for the LEFM has been provided for use in the uncertainty calculation described in Section I.1.E below.

I.1.D.v **Criterion 1 from ER-157P, Rev 8 - 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.**

A Selected Licensee Commitment (SLC) will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is not functional. This is identified as a Regulatory Commitment in Attachment 1. If a non-functional LEFM is not restored to functional status within 72 hours, then within 6 hours, the unit will be reduced to no more than 2568 MWt (the previously licensed rated thermal power).

The existing feedwater flow venturi-based signals will be corrected to the last valid data from the LEFM system during this period. Any slight drift of the feedwater flow venturi measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. This is conservative since the reactor will be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely and any significant sudden de-fouling would be detected by other plant parameters. It is expected that most issues rendering an LEFM System non-functional could be resolved within a 72-hour AOT. The NRC has approved a 72-hour AOT for previous Duke Energy MUR power uprate applications (McGuire Units 1 and 2 and Catawba Unit 1).

I.1.D.vi **Criterion 2 from ER-157P, Rev 8 - 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.**

ONS will not consider a CheckPlus system with a single failure as a separate category; such a failure will be considered as a non-functional LEFM and the same actions identified in response to Criterion 1 from ER-157P, Rev. 8 above will be implemented.

I.1.D.vii **Criterion 3 from ER-157P, Rev 8 - An applicant with a comparable geometry can reference the above Section 3.2.1 finding 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 the 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.**

As stated in response to Criterion 2 from ER-157P, Rev. 8 above, ONS will not consider a CheckPlus system with disabled components as a separate category; such a condition will be considered as an inoperable LEFM and the same actions identified in response to Criterion 1 above will be implemented.

I.1.D.viii **Criterion 4 from ER-157P, Rev 8 - An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 (Reference 17 = Letter from Hauser, E (Cameron Measurement Systems), to U.S. Nuclear Regulatory Commission, "Documentation to support the review of ER-157P, Revision 8: Engineering Report ER-790, Revision 1, 'An Evaluation of the Impact of 55 Tube Permutit Flow Conditioners on the Meter Factor of an LEFM CheckPlus'," March 19, 2010) 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.**

The ONS units have no flow straightener upstream (or downstream) of the LEFM installation. Therefore, this criterion is not applicable to ONS.

I.1.D.ix **Criterion 5 from ER-157P, Rev 8 - 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. (Reference 18 = Letter from Hauser, E (Cameron Measurement Systems), to U.S. Nuclear Regulatory Commission, "Documentation to support the review of ER-157P, Revision 8: Engineering Report ER-764, Revision 0, 'The Effect of the Distribution of the Uncertainty in Steam Moisture Content on the Total Uncertainty in Thermal Power'," March 18, 2010)**

The ONS Nuclear Steam Supply Systems (NSSSs) use Once-Through Steam Generators (OTSGs) that produce superheated steam as shown in Table IV-1 below. Thus, uncertainty associated with the steam moisture content at ONS is not a factor in the heat balance uncertainty calculation. This criterion is not applicable to ONS.

I.1.E **A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty**

RESPONSE:

Cameron calculations of LEFM uncertainty have been completed for each ONS unit. The calculations are listed below and are included in Attachments 6 (non-proprietary) and 8 (proprietary) to this License Amendment Request. Acceptance testing following installation of the Cameron CheckPlus systems in the ONS units will confirm that as built parameters are within the bounds of the error analyses.

The table below summarizes the instrument channel uncertainties used to determine the secondary power uncertainty while in "Normal" mode (no instrument failures). Note that two pressure and temperature instruments are available for each channel. These uncertainties combine to give an overall secondary heat balance power measurement uncertainty of 0.34% RTP.

Table I.1.E-1: Total Thermal Power Uncertainty Determination

Parameter	Uncertainty	Power Uncertainty
LEFM Power	0.30 %	0.30 % RTP
Feedwater Pressure	2.13 psi	0.0001 % RTP
Steam Enthalpy		
Temperature	1.77 °F	0.18 % RTP
Pressure	1.78 psi	0.02 % RTP
RCP, Makeup/Letdown Power	0.11 %	0.11 % RTP

I.1.F Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

RESPONSE:

I.1.F.i Maintaining calibration

Calibration of the LEFM will be ensured by preventative maintenance activities previously described in Section I.1.D.i, Response to Criterion 1 of ER-80P.

I.1.F.ii Controlling software and hardware configuration

The Cameron LEFM CheckPlus Systems were procured to the requirements of ANSI Std 7-4.3.2-2003 (Reference I.11) and ASME NQA-1, 2008 (Reference I.12). Hardware configuration will be controlled in accordance with Duke Energy procedures.

LEFM software will be classified in accordance with Duke Energy procedures. Software will be classified, developed, tested, and controlled in accordance with Duke Energy procedures. Implementation of the software will be performed under the design control process.

Instruments that affect the power calorimetric, including the Cameron LEFM CheckPlus System inputs, are monitored by ONS personnel. Equipment problems for plant systems, including the Cameron LEFM CheckPlus System equipment, fall under site work control processes. Conditions that are adverse to quality are documented under the corrective action program. Corrective action directives, which ensure compliance with the requirements of 10 CFR 50, Appendix B, include instructions for notification of deficiencies and error reporting.

I.1.F.iii Performing corrective actions

Corrective actions will be monitored and performed in accordance with Duke Energy procedures and the Work Process Manual.

I.1.F.iv Reporting deficiencies to the manufacturer

Reporting deficiencies to the manufacturer will be performed in accordance with Duke Energy procedures.

I.1.F.v Receiving and addressing manufacturer deficiency reports

Manufacturer deficiency reports will be received and addressed in accordance with Duke Energy procedures.

I.1.G A proposed allowed outage time for the instrument, along with the technical basis for the time selected

RESPONSE:

Refer to the response to I.1.D.v, Criterion 1 from ER-157P above.

I.1.H Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

RESPONSE:

The proposed actions to reduce power are stated in response to I.1.D.v, Criterion 1 from ER-157P above.

References for Section I:

I.1. Regulatory issue summary, RIS 2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002

I.2. Caldon, Inc., Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√ ™System," Revision 0, March 1997

I.3. Caldon® Ultrasonics Engineering Report: ER-157(P-A) Rev. 8 and Rev. 8 Errata, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," May 2008

I.4. NRC letter from John N. Hannon, 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)," March 8, 1999

I.5. NRC letter from Thomas B. Blount, Deputy Director, NRC, to Mr. 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)," August 16, 2010

I.6. Caldon® Ultrasonics Engineering Report: ER-855NP Rev 0, "Meter Factor Calculation and Accuracy Assessments for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3," November 2019

I.7. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation, dated September 20, 2011, License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02 (ML11269A127)

I.8. Letter to the U. S. Nuclear Regulatory Commission from T. Preston Gillespie, Jr., Vice President, Oconee Nuclear Station, Duke Energy Corporation, dated March 16, 2012, Supplement to License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02, Supplement 2 (ML12109A345)

I.9. American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1, "Measurement Uncertainty," 1985

I.10. ISA-RP 67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation," Approved September 1994

I.11. ANSI Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations – Annex E"

I.12. ASME NQA-1, 2008, "Quality Assurance Requirements for Nuclear Facility Applications"

I.13. Caldon® Ultrasonics Engineering Report: ER-813NP, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus™ System," November 2019 (Proprietary Version)

I.14. Caldon® Ultrasonics Engineering Report: ER-813P, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus™ System," November 2019 (Non-Proprietary Version)

I.15. Caldon® Ultrasonics Engineering Report: ER-824NP, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus™ System," November 2019 (Proprietary Version)

I.16. Caldon® Ultrasonics Engineering Report: ER-824P, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus™ System," November 2019 (Non-Proprietary Version)

I.17. Caldon® Ultrasonics Engineering Report: ER-825NP, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus™ System," November 2019 (Proprietary Version)

I.18. Caldon® Ultrasonics Engineering Report: ER-825P, Rev 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus™ System," November 2019 (Non-Proprietary Version)

I.19. Caldon® Ultrasonics Engineering Report: ER-972, "Traceability Between Topical Report (ER-157P-A Rev. 8 & Rev. 8 Errata) and System Uncertainty Report," Revision 3, September 2019

I.20. Caldon® Ultrasonics Engineering Report: ER-755, Revision 0, "Review of LEFM Impact on Venturi Discharge Coefficient," July 2009

I.21. Caldon® Ultrasonics Engineering Report: ER-855P Rev 0, "Meter Factor Calculation and Accuracy Assessments for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3," November 2019

II Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level

II.1 **A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):**

II.1.A **Identify the transient or accident that is the subject of the analysis**

II.1.B **Confirm and explicitly state that**

II.1.B.i **The requested uprate in power level continues to be bounded by the existing analyses of record for the plant**

II.1.B.ii **The analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC**

II.1.C **Confirm that bounding event determinations continue to be valid**

II.1.D **Provide a reference to the NRC's previous approvals discussed in Item B above.**

RESPONSE:

The response to II.1 is provided in Table II.1-1 – Oconee Analyses. Each analysis is described briefly below and all analyses are summarized in Table II.1-1, including the assumed core power level in each analysis and whether the analysis remains bounding for the MUR power uprate. The methodology in these analyses is found in Duke Energy Topical Reports, Vendor Topical Reports, and other reports as referenced in Table II.1-1. NRC review and approval of the applicable report is also referenced in Table II.1-1.

II.1.D.i Reactor Protection System Trip Function Allowable Values

The current safety analysis setpoint method is described in Chapter 4 of Reference II.1 The Technical Specification Allowable Values (AV) for current operation were used as the starting point for input to the safety analyses. The AV of interest for the MUR power uprate is the nuclear overpower (also known as high flux) trip function. In the safety analyses, the trip setpoint is identical to the AV.

The high flux trip setpoint AV for 4 Reactor Coolant Pump (RCP) operation is currently 105.5% of 2568 MWt (2709.2 MWt). All the UFSAR safety analyses were revised for the MUR power uprate by retaining the initial margin to the high flux trip AV. The high flux trip setpoint assumed in the safety analyses was increased to 107.5% of 2568 (2760.6 MWt). Following the uprate, the 105.5% RTP AV setpoint will be retained such that the new Technical Specification AV will be 105.5% of 2610 (2753.6 MWt). Since the safety analyses assume a high flux trip setpoint higher than the proposed AV (2760.6 vs. 2753.6 MWt), the safety analyses conservatively delay reactor trip on high flux. Following NRC approval, Duke Energy intends to use the proposed trip setpoint (2753.6 MWt) whenever a particular analysis is revised.

A high flux trip setpoint for 3 Reactor Coolant Pump (RCP) operation was approved by the NRC in Reference II.20. The high flux trip AV for 3 Reactor Coolant Pump (RCP) operation is currently 80.5% of 2568 MWt (2067.24 MWt), as approved by the NRC in Reference II.20. The current 3 RCP setpoint maintains the 4 RCP difference between rated thermal power and the high flux trip setpoint, i.e., 5.5%FP, as 3 RCP operation is limited to 75%FP. The nominal 3 RCP operation power level assumed in the safety analyses is unchanged for the MUR power uprate. The current nominal power level is 75% of 2568 MWt, which will become 73.8% of 2610 MWt, or 1926 MWt. Adding 5.5% to 73.8% yields the proposed 3 RCP high flux trip AV setpoint of 79.3% of 2610 MWt, or 2069.7 MWt. The basis for the 3 RCP high flux trip setpoint is to better mitigate the UFSAR Chapter 15.17 Small Steam Line Break transient initiated from 75% of 2568 MWt with 3 RCPs in operation. See the small steam line break discussion below.

If the LEFM is out of service for longer than the Selected Licensee Commitments (SLC) allowance, the high flux trip setpoint is returned to the pre-MUR power uprate value of 2709.2 MWt (103.8% of 2610 MWt). For 3 RCP operation, the high flux trip setpoint is reduced by 1.7% (1.64% rounded up), from 79.3% to 77.6% of 2610 MWt. The SLC will also suspend operations involving increasing power when above 2568 MWt. This maintains the analytical margin used in the safety analyses upon actuation of the high flux trip.

II.1.D.ii DNB Analyses in UFSAR Chapter 15

There are several analyses in the ONS UFSAR Chapter 15 that are evaluated for Departure from Nucleate Boiling Ratio (DNBR) concerns. These are described in ONS UFSAR Sections 15.2, 15.3, 15.5, 15.6, 15.7, 15.12, 15.13, and 15.17. These DNB events are evaluated using Duke Energy's Statistical Core Design (SCD) methodology. The treatment of core power initial conditions for SCD analyses are described in Section 4.1 of DPC-NE-3005-PA (Reference II.1). Appendix F of DPC-NE-2005-PA (Reference II.22, as approved by the NRC in Reference II.5) describes how a core power uncertainty of 2% is combined with other statistically treated uncertainties when developing the statistical DNBR design limit used in SCD analyses for the Mark-B-HTP fuel product at ONS. No credit has been taken for reduced core power uncertainty due to installation of the LEFMs when establishing the statistical DNBR design limit for SCD analyses. Therefore, the currently used statistical DNBR design limit for SCD analyses remains conservative for MUR conditions.

II.1.D.iii Discussion of RIS 2002-03 Section II.1 Events

1. Methodology (UFSAR Section 15.1)

   Section 15.1 of the UFSAR addresses the methodology used for the following sections.

2. Startup Accident – UFSAR Section 15.2

   The accident is initiated from beginning-of-cycle, 0 MWt and analyzed for peak primary pressure concerns. The reactivity insertion (uncontrolled control rod bank withdrawal) results in a rapid power excursion and corresponding heatup and pressurization. The accident analysis credits the high pressure, high flux, and flux/flow Reactor Protection System (RPS) trip functions. Since this accident is initiated from zero power, the MUR power uprate does not affect the analysis except as it affects the high flux and flux/flow trip setpoints. The startup accident currently summarized in Section 15.2 of the UFSAR used flux related trip setpoints that bound the MUR power uprate; and the resulting peak primary pressure is acceptable.

The analysis of record (AOR) for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

3. Rod Withdrawal at Power Accident – UFSAR Section 15.3

   The accident is initiated from beginning-of-cycle, hot full power and analyzed for departure from nucleate boiling (DNB) and peak primary pressure. The reactivity insertion (uncontrolled control rod group withdrawal) results in a power excursion and corresponding heatup and pressurization. The accident analysis credits the high pressure, high temperature, and high flux RPS trip functions. The rod withdrawal at power accident currently summarized in Section 15.3 of the UFSAR has been analyzed assuming an initial power level of 2619 MWt and a high flux trip setpoint which are bounding for the MUR power uprate; and the resulting DNB and pressure are acceptable.

   The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

4. Moderator Dilution Accident – UFSAR Section 15.4

   The moderator dilution accidents are initiated from both a full power and refueling condition. The analysis verifies there is at least 15 minutes (for the full power analysis) and 30 minutes (for the refueling mode analysis) for the operators to stop the dilution in time to prevent a return to criticality following a valid indication of a dilution.

   The initial and final boron concentrations are checked for each core reload design to ensure sufficient time exists to stop the dilution prior to the reactor returning to critical. Since each core design is performed at the rated thermal power for that core, this check will verify the MUR uprated core designs are acceptable with respect to this accident.

   The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

5. Cold Water Accident – UFSAR Section 15.5

   The cold water accident is analyzed with 3 reactor coolant pumps (RCPs) initially operating at a power level of 80% of 2568 MWt (2054 MWt). The analysis assumes the fourth RCP starts, which results in a power excursion due to the increased core flow. The event is analyzed to ensure acceptable DNB and peak RCS pressure results are obtained. The heat flux increases and attains a new steady-state power level but remains below 100% of 2568 MWt at all times which ensures the acceptance criteria are met. The proposed change to Technical Specification 3.4.4 will limit 3 RCP operation to 73.8% of 2610 MWt (1926 MWt), ensuring the initial power level assumed in the analysis is bounding. With this Technical Specification change, the MUR uprate will not invalidate the transient response or results of the current analysis.

   The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

6. Loss of Coolant Flow

6.a. Loss of Coolant Flow – Flow Coastdown – UFSAR Section 15.6

The various flow coastdown events analyzed in the UFSAR are a 4 RCP coastdown from 4 RCPs initially operating, 2 RCP coastdown from 4 RCPs, 1 RCP coastdown from 4 RCPs, 3 RCP coastdown from 3 RCPs and 1 RCP coastdown from 3 RCPs. The loss of a RCP, or multiple RCPs, leads to a reactor trip on either flux/flow or power/pump monitor, depending on the number of RCPs lost.

The analyses are performed for DNB concerns using the Statistical Core Design (SCD) methodology, and the most limiting of the flow coastdown events (2 RCP coastdown from 4 RCPs) is verified in the reload analyses. The current 2 pump coastdown analysis is initiated from 102% of 2568 MWt (2619 MWt), which bounds the power after the MUR uprate (2610 MWt). The flux/flow trip function will be recalibrated such that the flux part of the trip setpoint will not change for the MUR power uprate. Therefore, the current analysis results remain acceptable after the MUR uprate.

The current 3 RCP analyses summarized in UFSAR Section 15.6 have been performed at 80% of 2568 MWt (2054 MWt). The maximum allowed operating power for 3 RCPs (Technical Specification 3.4.4) will remain at 1926 MWt (now 73.8% of 2610) following the MUR uprate. The flux/flow trip function will be recalibrated such that the flux part of the trip setpoint will not change for the MUR power uprate. Therefore, the current 3 RCP analyses bound the MUR uprated power and the DNB results remain conservative.

There is a natural circulation capability analysis included in UFSAR Section 15.6.7 that demonstrates the successful establishment of natural circulation for a range of decay heat power. The MUR power uprate does not change the results and conclusions of the analysis but will result in a rescaling of the percent power documented in the UFSAR.

The AOR for this analysis is reflected in the Oconee UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

6.b. Loss of Coolant Flow – Locked Rotor – UFSAR Section 15.6

The locked rotor event is analyzed for a locked RCP rotor from full power with 4 RCPs initially operating and from 75% of 2568 MWt with 3 RCPs initially operating. The flow coastdown leads to a flux/flow trip and DNBR is the acceptance criterion evaluated using the SCD methodology.

The current 4 RCP analysis summarized in UFSAR Section 15.6 has been performed at 102% of 2568 MWt, which bounds the MUR uprate power level. The flux/flow trip function will be recalibrated such that the flux part of the trip setpoint will not change for the MUR uprate. Therefore, the current 4 RCP analysis results remain bounding after the MUR power uprate.

The current 3 RCP analysis summarized in UFSAR Section 15.6 has been performed at 75% of 2568 MWt (1926 MWt). The maximum allowed operating power for 3 RCPs (Technical Specification 3.4.4) will remain at 1926 MWt (73.8% of 2610) following the MUR uprate. The flux/flow trip function will be recalibrated such that the flux part of the trip setpoint will not change for the MUR uprate. Therefore, the current 3 RCP analysis results remain valid after the MUR uprate.

The AOR for this analysis is reflected in the Oconee UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

7. Control Rod Misalignment Accident – UFSAR Section 15.7

   There are three types of misalignments addressed in the UFSAR. They are statically misaligned rod, stuck rod, and dropped rod. The statically misaligned rod occurs when a control rod assembly is in motion and stops while the remaining assemblies in that group continue. It is analyzed for each reload core design to ensure the resultant core power distribution, after control rod motion stops, is acceptable. Since the analysis is performed for each reload, it will inherently include the MUR power uprate for cores that will operate at MUR uprated conditions.

   The stuck rod analysis is performed for each reload core design to ensure the core can maintain 1% Δk/k shutdown margin at hot shutdown conditions with the worst rod stuck in a fully withdrawn position. Since the analysis is performed for each reload, it will inherently include the MUR power uprate for cores that will operate at an MUR uprated condition.

   The dropped rod accident is analyzed with 4 RCPs from 102% of 2568 MWt (2619 MWt) and with 3 RCPs from 75% of 2568 MWt (1926 MWt). The control rod drops resulting in an initial decrease in power and a core wide tilt. The Integrated Control System (ICS) detects the decrease in power and pulls control rods to restore power back to the initial value. Reactor power increases due to feedback effects and control rod withdrawal, and a new steady-state power is achieved higher than the initial power level. The analysis is performed for DNB and Centerline Fuel Melt (CFM) concerns. With the proposed change to the Technical Specification power level for 3 RCP operation, the 3 RCP cases are unaffected. Since the 4 RCP case was initiated at a power level that bounds the MUR power uprate, the DNB and CFM results remain acceptable.

   The analyses of record for this classification of accidents are reflected in the ONS UFSAR and remain acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

8. Turbine Trip – UFSAR Section 15.8

   The turbine trip event is analyzed for 4 RCP operation at 102% of 2568 MWt and for 3 RCP operation at 82% of 2568 MWt. The acceptance criterion is peak primary pressure. The turbine trip is analyzed without credit for anticipatory reactor trip on turbine trip. The turbine trip causes an increase in secondary pressures and temperatures and reduces the primary-to-secondary heat transfer. The primary system heats up and pressurizes resulting in a high RCS pressure reactor trip. The MUR power uprate affects the analysis only as it affects the initial power level. Since the 4 RCP analysis assumed an initial power level that bounds the uprated power level, the 4 RCP analysis remains bounding after the MUR power uprate. With the proposed change to the Technical Specification power level for 3 RCP operation, the 3 RCP cases are unaffected.

   The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

9. Steam Generator Tube Rupture Accident – UFSAR Section 15.9

   The steam generator tube rupture accident is analyzed for 4 RCP operation at 102% of 2568 MWt. A double-ended guillotine tube rupture is postulated and operator actions are conservatively

modeled to depressurize and cool down the RCS until primary-to-secondary break flow and steam releases are terminated. The MUR power uprate affects the analysis only as it affects the initial stored energy and subsequent cool down. The acceptance criterion is offsite dose remaining less than applicable regulatory limits (currently, 100% of 10 CFR part 100 limits).

The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

The radiological dose analysis for the steam generator tube rupture accident uses a bounding fission product inventory based on operation at 102% of 2568 MWt (2619 MWt) which bounds the MUR uprate power level. Consequently, the steam generator tube rupture radiological analysis remains acceptable for the MUR power uprate.

10. Waste Gas Tank Rupture Accident – UFSAR Section 15.10

The accident summarized in UFSAR Section 15.10 is the rupture of a waste gas tank resulting in the release of the radioactive contents of the tank to the plant auxiliary building ventilation system and to the atmosphere through the unit vent. The acceptance criterion is that the dose at the exclusion area boundary (EAB) remains less than the Technical Specification 5.5.13 limit of 500 mrem.

The radiological dose analysis for the waste gas tank rupture uses a bounding fission product inventory based on operation at 102% of 2568 MWt (2619 MWt).

The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate.

11. Fuel Handling Accidents – UFSAR Section 15.11

The fuel handling accidents include four base accidents, with one base accident having three separate considerations.

1. Base Case Fuel Handling Accident in Spent Fuel Pool (UFSAR Section 15.11.2.1)
2. Base Case Fuel Handling Accident Inside Containment (UFSAR Section 15.11.2.2)
3. Fuel Shipping Cask Drop Accidents (UFSAR Section 15.11.2.4)
4. Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building (UFSAR Section 15.11.2.5)
   - Criticality Analyses for Dry Storage Transfer Cask Drop Scenarios
   - Potential Damage to SFP Structures from Dry Storage Transfer Cask Drop
   - Radiological Dose from Dry Storage Transfer Cask Drop

Each of these fuel handling radiological dose analyses use a bounding fission product inventory, based on operation at 102% of 2568 MWt (2619 MWt) which bounds the MUR uprate power level. Consequently, the fuel handling accident analyses remain acceptable for the MUR power uprate. The source term and calculated dose results for the fuel handling accidents were reviewed and

approved as part of the License Amendment Request for full-scope implementation of the Alternative Source Term (Reference II.9).

12. Rod Ejection Accident – UFSAR Section 15.12

The rod ejection accident is analyzed at hot zero power (HZP), at 77% of 2568 MWt with 3 RCPs in operation, and at 102% of 2568 MWt with 4 RCPs in operation, all at beginning-of-cycle and end-of-cycle conditions. The analyses are performed for peak fuel rod enthalpy, DNBR, peak RCS pressure, and to generate thermal-hydraulic input to the dose analysis. A conservatively large rod worth is ejected causing a rapid power excursion. The event is mitigated first by Doppler feedback then by control rod insertion as reactor trip occurs on high flux. All analyses are performed assuming a high flux trip setpoint corresponding to the design overpower value of 112% of 2568 MWt, which is considerably higher than the proposed high flux trip setpoint.

The HZP analysis results are unaffected by the MUR power uprate. Since the 3 RCP cases were initiated from a power level (1977.4 MWt) that bounds the allowed power level for 3 RCP operation (1926 MWt), and since the 4 RCP cases were initiated from a power level (2619 MWt) that bounds the MUR power uprate power level, all DNB, peak enthalpy, and peak RCS pressure results are acceptable for a MUR power uprate.

A previous LAR related to use of Gadolinia as an integral burnable absorber in ONS cores (References II.25 and II.26) has been approved by the NRC in Reference II.21. The Gadolinia LAR revised the rod ejection accident method by crediting the flux/flow trip function for the 3 RCP and HZP analyses. As explained above in the loss of flow accident, the flux/flow trip setpoint is not changing for the MUR power uprate. Consequently, the method submitted in References II.25 and II.26, and the analyses performed with that method remain valid for the MUR power uprate.

The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

The radiological dose analysis for the rod ejection accident uses a bounding fission product inventory, based on operation at 102% of 2568 MWt full power conditions (2619.4 MWt). The analysis methodology utilized is the Alternative Source Term approved by the NRC in Reference II.19. Consequently, the rod ejection dose analysis remains acceptable for the MUR power uprate.

13. Steam Line Break Accident– UFSAR Sections 15.13 and 5.2.3.4

All the UFSAR Chapter 15 analyses were revised in anticipation of the MUR power uprate, including the MSLB analysis. At the time the MSLB analyses were performed, the magnitude of the MUR power uprate was unknown, therefore to bound all potential uprate conditions the MSLB analyses were performed at 102% of 2568 MWth, or 2619 MWth.

For MSLB with offsite power available, Section 15.3.1.1.1 of DPC-NE-3005-PA (Reference II.1) states that full rated power plus uncertainty is assumed for the initial power level. Therefore, the MSLB with offsite power analyses described in the UFSAR Section 15.13 analysis remains acceptable for the MUR power uprate.

For MSLB without offsite power available, Section 15.3.2.1.1 of DPC-NE-3005-PA states that nominal full power initial condition is assumed since the uncertainty in power is accounted for in the Statistical Core Design (SCD) limit. The treatment of core power initial conditions for SCD analyses are described in Section 4.1 of DPC-NE-3005-PA. Appendix F of DPC-NE-2005-PA (Reference

II.22, as approved by the NRC in Reference II.5) describes how a core power uncertainty of 2% is combined with other statistically treated uncertainties when developing the statistical DNBR design limit used in SCD analyses for the Mark-B-HTP fuel product at ONS. No credit has been taken for reduced core power uncertainty due to installation of the LEFMs when establishing the statistical DNBR design limit for SCD analyses. Therefore, the currently used statistical DNBR design limit for SCD analyses remains conservative for MUR power uprate conditions, and the MSLB without offsite power analyses described in UFSAR Section 15.13 analysis remains acceptable for the MUR power uprate.

The MSLB analysis postulates a double-ended rupture of the main steam line from one SG upstream of the turbine stop valves. The break results in a rapid overcooling and depressurization of the primary system. At end-of-cycle conditions with a large negative moderator temperature coefficient (MTC), the overcooling could lead to a return to power following reactor trip. The combination of low flow, low pressure, and a potential return to power leads to DNBR concerns. The MSLB analysis is also performed to quantify steam release through the break for input to the dose analysis. The MUR power uprate will not affect the DNBR analysis as the initial power level remains bounding. The MUR power uprate will not affect the dose analysis since the initial power level, the stored energy in the SSCs, and the decay heat following operation at 102% of 2568 MWt remain bounding after the MUR power uprate.

The radiological dose analyses for the steam line break accident used a bounding fission product inventory, based on extended operation at 102% of 2568 MWt full power conditions (2619.4 MWt). Consequently, the MSLB radiological analysis remains acceptable for the MUR uprated core.

The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

The MSLB analysis is also analyzed for steam generator tube integrity as described in UFSAR Section 5.2.3.4. The thermal-hydraulic input to the tube stress analysis was first performed with RETRAN-02 (References II.31, II.32, and II.33). The analysis was reviewed and approved by the NRC in Reference II.27 and was initiated from 102% of 2568 MWt (2619 MWt). The RETRAN-02 analysis was subsequently replaced with a RETRAN-3D (Reference II.28) ROTSG analysis, which was also initiated from 102% of 2568 MWt. Since acceptable tube stresses were obtained for the analysis initiated at a power level that bounds the MUR power uprate, the analysis remains acceptable for the MUR power uprate.

MSLB is also analyzed to provide mass and energy (M&E) data for use in the containment analyses. More discussion is provided in the Containment Performance section (Item 20) below.

14. Loss of Coolant Accidents – UFSAR Sections 15.14 and 5.2.3.4

The loss of coolant accidents analyzed have been reviewed for the impact of the uprate. Based on the power levels assumed in the current safety analyses, it has been determined that all LOCA analyses bound the uprate. Since the proposed change relies on less than 0.4% uncertainty, the nominal power level of 100.34% of 2610 MWt reflects the analysis power of 2619 MWt, and all five criteria of 10 CFR 50.46 continue to be met following a LOCA initiated at the post-MUR power level. Therefore, LOCA analyses performed at this power remain bounding. The LOCA analyses described in UFSAR Section 15.14 were performed with NRC-approved methods described in Reference II.23.

The ROTSG design basis analyses include tube stresses resulting from a pressurizer surge line break and a hot leg break at the top of the "candy-cane." The stress results are then used to define an allowable tube flaw size to provide the basis for condition monitoring. These results are documented in UFSAR Section 5.2.3.4. The thermal-hydraulic input to the tube stress analysis was generated by Framatome (formerly AREVA) using their NRC approved RELAP5 model and was performed at a power level greater than 102% of 2568 MWt (Reference II.24). Consequently, the tube stresses following a pressurizer surge line break and hot leg break are acceptable for the MUR power uprate.

LOCAs are also analyzed to provide M&E data for use in the Containment analyses. More discussion is provided in the Containment Performance section (Item 20) below.

15. Maximum Hypothetical Accident – UFSAR Section 15.15

    The radiological dose analysis for the maximum hypothetical accident uses a bounding fission product inventory, based on operation at 102% of 2568 MWt (2619 MWt). The source term and calculated dose results for the maximum hypothetical accident were reviewed and approved as part of the License Amendment Request for full-scope implementation of the Alternative Source Term (Reference II.9). Consequently, the maximum hypothetical accident radiological analysis remains acceptable for a MUR power uprate.

16. Post Accident Hydrogen Control – UFSAR Section 15.16

    The analysis documented in the UFSAR is historical. The original intent was to demonstrate the hydrogen recombiners could successfully prevent the buildup of excessive hydrogen concentrations in containment following a design basis large break LOCA. Duke Energy submitted an LAR to remove the hydrogen recombiners from service and obtained NRC approval in Reference II.29. The Safety Evaluation (SE) in Reference II.29 was issued based on analyses performed with the NUREG 0800 Section 6.2.5 prescribed computer code COGAP (Combustible Gas Analyzer Program, Reference II.30). The analysis results demonstrated hydrogen concentrations generated following a LBLOCA were less than the lower flammability limit of 4 v/o for the first 15 days post-LOCA and a maximum of 6.4 v/o 30 days post-LOCA without the operation of the hydrogen recombiner system. The sensitivity studies that generated these results assumed an initial power level of 102% of 2568 MWt. Since the NRC conclusion to allow the removal of the hydrogen recombiners was based on analyses initiated from 102% of 2568 MWt, the NRC conclusion remains valid following an MUR power uprate.

17. Small Steam Line Break Accident – UFSAR Section 15.17

    The small steam line break accident is analyzed at 102% of 2568 MWt with 4 RCPs in operation for DNB, CFM, and to provide thermal-hydraulic and steam release input to the dose analysis. A small break of a steam line is postulated that causes an overcooling event that results in a new, elevated steady-state power level. For the current UFSAR analyses, the magnitude of the power excursion is limited by the high flux trip function (4 RCP case), while the break size is limited by the high flux trip and variable low pressure-temperature trip. Ten minutes following the break, manual operator action is credited for tripping the reactor. The dose input analysis then models the plant cool down to Decay Heat Removal (DHR) conditions and calculates the resultant steam release. Since the 4 RCP case was initiated from a power level (2619 MWt) that bounds the MUR uprate power level, all DNB, CFM, and steam release results are bounding for the MUR power uprate.

    A high flux trip setpoint for 3 RCP operation was approved by the NRC in Reference II.20. The basis for this high flux trip setpoint is to better mitigate the 3 RCP small steam line break. The small

steam line break accident is analyzed at 75% of 2568 MWt with 3 RCPs in operation for DNB concerns using the SCD methodology. The addition of the new 3 RCP high flux trip setpoint for the nuclear overpower trip function results in a much lower steady-state power level and improved DNB results for the 3 RCP small steam line break analysis.

The 3 RCP case was initiated from a power level of 75% of 2568 MWt, which equates to 73.8% of 2610 MWt, consistent with the proposed change to Technical Specification 3.4.4 for MUR power uprate conditions.

The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

The radiological dose analyses for the small steam line break use a bounding fission product inventory based on operation at 102% of 2568 MWt (2619 MWt). Consequently, the small steam line break radiological analysis remains acceptable for the MUR power uprate.

18. Anticipated Transient Without Scram (ATWS) – UFSAR Sections 15.18 and 7.8

    An ATWS is initiated from either a loss of main feedwater (LOMFW) or a loss of offsite power (LOOP) event from 102% of 2568 MWt to demonstrate the adequacy of the ATWS Mitigation System Actuation Circuitry (AMSAC) and Diverse Scram System (DSS) systems. The limiting condition and primary safety concern associated with these two transients is the potential for high pressure within the Reactor Coolant System (RCS). Neither of the initiating events is postulated to cause an automatic reactor trip (i.e., de-energization of the control rod drive mechanisms) thereby relying on AMSAC and DSS to trip the control rods into the core. The DSS setpoint is 2450 psig ± 25 psig. The acceptance criterion is peak RCS pressure less than 3250 psia. The system response to each initiating event has been analyzed using NRC reviewed and approved methods (Reference II.11). The results demonstrate that the AMSAC and DSS systems described in UFSAR Section 7.8 are sufficient to meet the acceptance criterion. The MUR power uprate does not affect the analysis since the initial power level, stored energy of the SSCs, and decay heat are representative of the MUR power uprate values for those parameters.

19. Natural Circulation Cooldown – UFSAR Section 5.1.2.4

    ONS developed a procedure to continuously vent the reactor vessel head to containment during a natural circulation cooldown to DHR System conditions in response to Generic Letter 81-21. The technical basis behind the procedure is a RETRAN-02 (References II.31, II.32, and II.33) analysis of the reactor vessel head and head vent flow path. The acceptance criterion of cooling down without voiding the reactor vessel head region was successfully demonstrated and the NRC accepted this response in Reference II.17. The MUR power uprate does not affect the analysis because decay heat is not explicitly modeled. The atmospheric dump valves (ADVs) and the Turbine Bypass System (TBS) are capable of accommodating the increased steam loads. Therefore, increased decay heat generation will not impact the cool down rate assumption in the analysis and, consequently, the analysis remains valid following an MUR power uprate.

    Furthermore, as discussed in the NFPA-805 Fire response (Item 24 below), a containment response analysis assuming a natural circulation cooldown is performed to generate the pressure/temperature profiles input to the Environmental Qualification (EQ) analyses. The M&E data used in the containment response analysis is generated by a RETRAN-02 analysis initiated from 2619 MWt which bounds the MUR power level. Therefore, the containment response is conservative relative to the MUR power uprate.

20. Containment Performance – UFSAR Section 6.2

Containment short term pressurization following a LOCA is discussed in UFSAR Section 6.2.1.1.3.1, containment long term temperature response following a LOCA is discussed in UFSAR Section 6.2.1.1.3.2, and containment temperature and pressure following a steam line break is discussed in UFSAR Section 6.2.1.1.3.3.

These analyses are performed to ensure the containment pressure limit is not exceeded and the temperature response assumed in the EQ analyses remain bounding. Additionally, small break LOCAs (SBLOCA) are analyzed to verify large break LOCA (LBLOCA) is more limiting. The LOCA and steam line break M&E release data input to the containment analyses are performed at 2619 MWt (102% of 2568). The peak containment pressure is below the design limit and the temperature profile assumed in the EQ analyses is not challenged.

The AOR for these analyses is reflected in the ONS UFSAR and remain acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

21. Environmental Qualification (EQ) Parameters

The pressure and temperature profiles generated by the various transient analyses are summarized here. The MUR power uprate affects the mass and energy release data input to the various structure analyses and consequently affects the EQ analyses. The containment response analyses following a LOCA or MSLB described previously (Item 20 above) all obtain the M&E data from analyses performed at 2619 MWt (102% of 2568 MWt). The containment response analysis following an NFPA-805 based fire (described below in Item 24) also obtains the M&E data from analyses performed at 2619 MWt.

The Penetration Room response analyses following a main feedwater line break (MFWLB) (large break and critical crack) or MSLB in the penetration room obtain the M&E data from analyses performed at 2619 MWt. The large break MFWLB penetration room pressure/temperature results bound the critical crack penetration room results. The large break MFWLB and MSLB M&E analyses are generated using the NRC reviewed and approved methods in Reference II.11 at 2619 MWt.

The M&E analyses are performed at an initial power level that bounds the MUR power uprate and result in a conservative pressure/temperature profile for use in the EQ analyses.

Dose input to the EQ analyses is addressed in Section V.1.C.

22. Flood

22.a. Standby Shutdown Facility (SSF) Event Turbine Building Flood

A reconstituted ONS Turbine Building Flood Mitigation Strategy was approved by the NRC in Reference II.36. Transient analyses in support of the Turbine Building Flood mitigation were performed using the RETRAN-3D computer code. The NRC-approved licensing basis for Turbine Building Flood events mitigated by the SSF includes events occurring when the ONS units are not at nominal full power conditions that allows use of off-nominal success criteria. Turbine Building Flood events initiated at full power conditions were analyzed with an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR power uprate conditions.

22.b. External Flood

Transient analyses in support of External Flood mitigation strategies due to a postulated Jocassee Dam failure were performed using the RELAP5/MOD2-B&W computer code. These analyses were evaluated as part of the mitigating strategies for Beyond-Design-Basis external events which credit use of FLEX equipment. The NRC Safety Evaluation for these mitigating strategies is documented in Reference II.42. These events were analyzed with an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR uprate conditions.

23. Station Blackout

Station Blackout (SBO) is the hypothetical case where all off-site power and both Keowee hydro-electric units are lost. Electrical power is available immediately from the battery systems and within ten minutes from the SSF diesel generator. The MUR power uprate will have no impact on the design of or the loads supplied from both the battery systems and the SSF diesel generator. Therefore, capacity and capability of electrical power systems for SBO event for plant operation under MUR power uprate conditions are bounded by the load profiles, which are supported by the existing analysis of record. As a result, the SSF will continue to have adequate capacity and capability to operate the plant equipment. The current analysis of record, UFSAR Section 8.3.2.2.4, remains bounding for the MUR power uprate.

This event was originally included in UFSAR Section 15.8.3. As documented in the NRC Safety Evaluation Report (SER) dated March 10, 1992 and the NRC Supplemental SER dated December 3, 1992, ONS complies with 10 CFR 50.63 and conforms to the guidance of NUMARC Report 8700 and Regulatory Guide 1.155. This regulation requires that a licensed nuclear power plant demonstrate the ability to achieve safe shutdown from 100% reactor power by ensuring containment integrity and adequate decay heat removal for a calculated duration. As discussed in Section V.1.B below, ONS Units 1, 2, and 3 will still be able to achieve safe shutdown following the MUR power uprate.

24. NFPA-805 Fire

ONS has transitioned from an Appendix R licensing basis fire analysis to the probabilistic NFPA-805 fire analysis in accordance with Reference II.34. Full implementation of the Protected Service Water System was completed per Reference II.37. The supporting thermal-hydraulic analyses are initiated from 2619 MWt (102% of 2568). The analyses successfully demonstrate that both shutdown margin and natural circulation are maintained. Ambient heat loss from the RCS and thermal-hydraulic inputs from this event are input to a containment analysis to demonstrate containment pressure and temperature are within the bounds of the EQ of the equipment relied upon to mitigate this event.

The analysis is performed using the NRC reviewed and approved methods (Reference II.11). Since the event is analyzed at an initial power level that bounds the MUR uprated power, the NFPA-805 based fire results are acceptable for the MUR power uprate.

The natural circulation cooldown analysis is performed to provide input to a containment analysis. The containment analysis provides a pressure/temperature profile validated by the EQ analysis which verifies operability of the equipment in containment relied upon for mitigation of this event. The analysis is initiated from 2619 MWt (102% of 2568). The results demonstrate that natural circulation is successfully established and maintained for the safe shutdown condition.

25. Spent Fuel Pool Accidents (loss of pool cooling)

    This is neither a design basis event nor a scoping event for ONS. It is not a Chapter 15 event nor a natural phenomenon event. A loss of SFP cooling is assumed to occur as part of the Turbine Building flooding scenario. The SFP inventory analysis was reviewed to confirm that the stored fuel remains covered during this event. The calculation is based on the total allowable spent fuel pool heat load and not just on the operating power level. Loss of SFP cooling was also postulated to occur at other times to establish heat up rates to determine how much time it took before onset of boiling. Adequacy of time for corrective action is the docketed success criterion. No imposition of design requirements resulted. This event is only used to assess performance of the SFP cooling system.

26. Loss of Main Feedwater (UFSAR Section 10.4.7.3.1)

    The loss of main feedwater event (LOMFW) is analyzed to demonstrate the adequacy of the Emergency Feedwater (EFW) System. It is initiated from 102% of 2568 MWt and is analyzed for peak RCS pressure. The MUR power uprate does not affect the analysis since the initial power level, stored energy of the SSCs, and decay heat values used are the result of operation at 102% of 2568 MWt. The LOMFW is analyzed using the NRC reviewed and approved RETRAN-3D transient analysis computer code.

    The AOR for this analysis is reflected in the ONS UFSAR and remains acceptable for the MUR power uprate. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

27. Feedwater Line Break - UFSAR Sections 3.6 and 10.4.7.3.

    Feedwater Line Breaks are evaluated as part of the High Energy Line Break (HELB) program. The current licensing basis for Feedwater Line Break analyses are bounded by the transient analyses performed for Feedwater Line Breaks in support of the future licensing basis for HELB, described below.

    A reconstituted Oconee HELB Mitigation Strategy License Amendment Request (LAR) has been submitted to the NRC in Reference II.38. Transient analyses for Feedwater Line Breaks in support of the HELB LAR were performed using the RELAP5/MOD2-B&W computer code using the NRC reviewed and approved methods (Reference II.19). Attachment 4 of Reference II.38 describes Oconee RELAP5 model modifications made relative to the methods described in Reference II.19, which were developed specifically to perform overheating and overcooling transient analyses of HELB scenarios. The full power Feedwater Line Break analyses assumed an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR uprate conditions. Additional information regarding the suitability of RELAP5/MOD2-B&W for use in ONS non-LOCA overheating analyses was provided to the NRC in Reference II.39. As documented in Reference II.43, the NRC has approved the use of RELAP5/MOD2-B&W for ONS non-LOCA overheating events, such as feedwater line breaks.

    FWLB cases are also analyzed to provide M&E input used to generate the pressure and temperature profiles used in the EQ analyses. See the EQ parameters discussion in Item 21 above.

28. LTOP

    Low-Temperature Overpressure Protection is discussed in Section IV.1.C.v.

29. High Energy Line Break (HELB)

The current licensing basis for HELB analyses are bounded by the transient analyses performed in support of the future licensing basis for HELB, described below.

A reconstituted Oconee HELB Mitigation Strategy License Amendment Request (LAR) has been submitted to the NRC in Reference II.38. Transient analyses in support of the HELB LAR were performed using the RELAP5/MOD2-B&W computer code, using the NRC reviewed and approved methods (Reference II.19). Attachment 4 of Reference II.38 describes Oconee RELAP5 model modifications made relative to the methods described in Reference II.19, which were developed specifically to perform overheating and overcooling transient analyses of HELB scenarios. The HELB analyses assumed an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR power uprate conditions. Additional information regarding the suitability of RELAP5/MOD2-B&W for use in ONS non-LOCA overheating and overcooling analyses was provided to the NRC in Reference II.39.

30. RPS/ES Instrument Uncertainties - Calculation and Application

Uncertainty Calculation:

The RPS and Engineered Safeguards (ES) instrument uncertainty calculations are performed per Reference II.35. The MUR power uprate is accomplished by reducing the uncertainty in the secondary side heat balance. As a result of the MUR power uprate, full power is increased and consequently any component expressed as a percent of full power span is also potentially impacted. Therefore, the MUR power uprate potentially affects those RPS/ES instrument uncertainties that contain a term for the secondary side heat balance uncertainty and/or instrument string components sensitive to the full power span.

A review of the RPS/ES uncertainties reveals three uncertainty calculations are potentially impacted:

1. The nuclear overpower trip function (or high flux trip function),
2. The nuclear overpower flux/flow imbalance trip function (or flux/flow/imbalance trip function),
3. The reactor coolant pump to power trip function (or pump monitor trip function).

Neither the flux/flow/imbalance trip function uncertainty calculation nor the pump monitor trip function contains a heat balance term. Consequently, they are only potentially impacted by the percent full power span. The percent full power span will be retained and consequently, the uncertainty calculated is unaffected. Currently, the excore Nuclear Instrumentation (NI) detectors are calibrated to a span of 0-62.5% of 2568 MWt. For the MUR power uprate, they will be rescaled to 0-62.5% of 2610 MWt. Therefore, the flux/flow/imbalance and pump monitor uncertainty calculations are acceptable for the MUR power uprate.

The high flux trip function uncertainty calculation contains a heat balance term. The calculation has been revised for the reduced heat balance uncertainty to document a total loop uncertainty applicable to an MUR uprated core.

As described in Chapter 4 of DPC-NE-3005-PA (Reference II.1), the safety analyses assume the RPS trip setpoints are the Technical Specification allowable values given in Technical Specification Table 3.3.1-1. The safety analyses then error adjusts the signal being compared to the setpoint such that conservatism is introduced in the parameter(s) actuating reactor trip.

The high flux trip analytical limit in the safety analyses accounts for two different uncertainty terms. One is the core thermal power measurement uncertainty in which feedwater flow measurement is the most significant contributor. The other uncertainty term is associated with the Power Range Nuclear Instrumentation (NI). Guidance exists in plant operating procedures that does not allow NI indications to be greater than 2% non-conservative during unit operation. Both the NI uncertainty and heat balance uncertainty are accounted for in the analytical limit in the safety analyses. When the LEFM is functional the core thermal power measurement uncertainty is 0.34%, which is incorporated in the post-MUR high flux analytical limits. When the LEFM is nonfunctional this uncertainty term increases back to 2%. This increase in feedwater flow measurement uncertainty, when the LEFM is nonfunctional, is not bounded by the analytical limit in the safety analyses. Therefore, the high flux trip and the flux/flow setpoints are being reduced to maintain the analytical limit.

When the UFSAR Chapter 15 safety analyses were performed for the MUR power uprate, the flux/flow/imbalance envelope was not adjusted for the increase in power. Since acceptable results are obtained, the flux/flow/imbalance envelope was not changed, and is considered acceptable at the current values. Since the flux/flow/imbalance envelope is not being increased for the uprate, at least initially, the initial margin to trip is decreased.

The LEFM uncertainty impact on the secondary power for 3 RCP operation is the same as the LEFM uncertainty impact on the secondary power for 4 RCP operation. Therefore, the calculated heat balance uncertainty is equally applicable to both the 4 RCP and 3 RCP high flux trip setpoint uncertainty calculations.

All other RPS/ES trip functions do not contain a heat balance component or percent full power span component in the instrument string and are consequently unaffected by the MUR power uprate.

Uncertainty Application:

The NRC approved safety analysis setpoint method documented in Chapter 4 of Reference II.1 describes how uncertainties related to the RPS trip functions are applied in the safety analyses. The approved method for accidents that trip on the high flux trip function algebraically sums the steady-state excore NI uncertainty, the heat balance uncertainty, and any transient NI effects specific to the transient being analyzed. In the safety analyses, when the excore NI signal reaches the high flux trip function allowable value, a reactor trip occurs after an appropriate delay. The analytical limit is then the algebraic sum of the three uncertainty terms added to the allowable value. The actual power at reactor trip would be less than or equal to the analytical limit for that transient. The MUR power uprate does not affect the Reference II.1 method, but does reduce the magnitude of the heat balance uncertainty.

Per Chapter 4 of Reference II.1, the NRC approved method for accidents that trip on the flux/flow/imbalance trip function treats the uncertainty in power the same way it is treated for those transients that trip on the high flux trip function. That is, it algebraically sums the steady-state excore NI uncertainty, the heat balance uncertainty, and any transient NI effects specific to the transient being analyzed. In the safety analyses, reactor trip occurs when the excore NI signal reaches the flux/flow trip function allowable value. The main difference from the high flux trip function is that the flux/flow trip setpoint is dynamically compensated for changes in RCS flow. The flow uncertainty is also modeled as it affects the trip setpoint. The analytical limit is then the algebraic sum of the three uncertainty terms added to the allowable value, which is adjusted for the flow uncertainty. The actual power at reactor trip would be less than or equal to the analytical limit for that transient. Like the high flux trip function, the MUR power uprate does not affect the Reference II.1 method but does reduce the magnitude of the heat balance uncertainty.

31. Natural Phenomena

Four separate natural phenomena analyses are scoping events for ONS: (1) tornado, wind, hurricane; (2), seismic; (3), external floods; (4), snow and ice. Natural phenomena events are not design basis events (DBEs) at ONS; instead, they impose design criteria on SSCs identified for mitigation of accidents.

A discussion of each event is presented below:

1) Tornado, wind and hurricane.

The tornado, wind, and hurricane analysis establishes design criteria for SSCs. As discussed in UFSAR Section 3.2.2, the Reactor Coolant System, by virtue of its location in the Reactor Building, will not be damaged by a tornado. Capability is provided to safely shutdown all three units. As part of the original FSAR development, specific accident analyses were not performed.

A reconstituted ONS Tornado Mitigation Strategy LAR was submitted to the NRC in Reference II.40. This LAR was approved by the NRC in an SE (Reference II.43). Transient analyses in support of the Tornado LAR were performed using the RELAP5/MOD2-B&W computer code, and those analyses assumed an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR power uprate conditions. Additional information regarding the suitability of RELAP5/MOD2-B&W for use in ONS non-LOCA overheating and overcooling analyses was provided to the NRC in Reference II.39. As documented in Reference II.43, the NRC has approved the use of RELAP5/MOD2-B&W for ONS non-LOCA overcooling and overheating events.

The reconstituted ONS Tornado Mitigation Strategy credits the SSF as the assured mitigation path following a tornado with the assumed initial conditions of loss of all AC power to all units with significant tornado damage to one unit. The SSF System is addressed in Sections IV.1.A.v, V.1.B, and VI.1.C.v.

2) Seismic events

The SSCs to be protected from the effects of natural phenomena are identified within the UFSAR and other licensing documents.

For ONS, the "earthquake event" is simply a set of forces and loads applied to certain specific SSCs. The characteristics of those forces and loads were calculated and applied based on seismic analyses, but an actual earthquake, with all its potential effects, is not postulated.

Duke Energy submitted information explaining the seismic basis to the NRC in 1994 (Reference II.45). The NRC addressed that information in Reference II.46. The correspondence explains that, although seismic loads are used as design criteria for SSCs that mitigate and prevent the LBLOCA/LOOP, a seismic event or an independent pipe break is not postulated to occur concurrently with a LOCA. This means that certain SSCs used in the prevention and mitigation of a LBLOCA are designed using seismic loads, but the licensing basis is that the LOCA and seismic event do not occur simultaneously. The UFSAR accident analyses do not assume the seismic event and LOCA occur at the same time. The MUR power uprate does not affect the current seismic basis for the plant.

3) External floods

The systems and components to be protected from the effects of natural phenomena are identified within the UFSAR and other licensing documents. External floods impose design criteria on SSCs identified for mitigation of accidents. By letter dated January 31, 2017 (Reference II.47), Duke submitted its External Flood mitigation strategies assessment for ONS. This submittal was in response to NRC Order EA-12-049 (Reference II.48). Transient analyses in support of External Flood mitigation strategies due to a postulated Jocassee Dam failure were performed using the RELAP5/MOD2-B&W computer code. These events were analyzed with an initial reactor power of 102% of 2568 MWt, which bounds the ONS MUR power uprate conditions. The analysis credits the SSF and FLEX equipment for mitigation of an external flood event.

4) Snow and ice

Snow and ice establishes external loads for buildings and structures. The systems and components to be protected from the effects of natural phenomena are identified within the UFSAR and other licensing documents. Thus, there is no specific snow and ice analysis at ONS, and no specific power level is assumed. The MUR power uprate does not affect the current snow and ice basis for the plant.

32. Double Steam Line Break Mitigated by the Safe Shutdown Facility

The ONS SSF serves as a backup to the HPI and EFW systems in the case of failure of these systems for any reason, including sabotage events. A double steam line break (DSLB) was established as the worst-case event to be mitigated by the SSF. Duke Energy submitted a report to the NRC on July 9, 1986 (Reference II.41) which documented an analysis of a DSLB event and the effectiveness of the SSF in mitigating the RCS response to a DSLB. That DSLB analysis was performed at an initial reactor power level of 2568 MWt. The success criteria demonstrated in the original DSLB analysis were that the reactor never returns to criticality, the active fuel is not uncovered, and that long-term natural circulation is not halted.

Recent DSLB analyses crediting the SSF have been performed in support of the reconstituted ONS Tornado Mitigation Strategy (Reference II.40) and the reconstituted ONS High Energy Line Break (HELB) Mitigation Strategy (Reference II.38). These recent DSLB analyses were performed using the RELAP5/MOD2-B&W computer code, and assumed the initial reactor power was 102% of 2568 MWt, which bounds the ONS MUR power uprate conditions. These postulated DSLB scenarios mitigated by the SSF, as described in References II.38 and II.40, are evaluated for impacts to fuel integrity (e.g. Departure from Nucleate Boiling Ratio limits), RCS pressure and temperature limits, and demonstrate that the core remains intact and in a coolable geometry. Therefore, these analyses are used to demonstrate the SSF is capable of successfully mitigating a DSLB using the SSF. As documented in Reference II.43, the NRC has approved the use of RELAP5/MOD2-B&W for ONS non-LOCA overcooling events such as DSLB due to tornado events.

References for Section II:

II.1. DPC-NE-3005-PA, Revision 5, “Oconee Nuclear Station UFSAR Chapter 15 Transient Analysis Methodology,” April 2016

II.2. Letter from David LeBarge (NRC) to W. R. McCollum (Duke) dated October 1, 1998, “Review of Updated Final Safety Analysis Report, Chapter 15, Transient Analysis Methodology Submittal – Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. M99349, M99350, and M99351)”

II.3. Letter from David LeBarge (NRC) to W. R. McCollum (Duke) dated May 25, 1999, “Oconee Nuclear Station, Units 1, 2, and 3 Re: Safety Evaluation for Revision 1 to Topical Report DPC-NE-3005-P, “UFSAR Chapter 15 Transient and Accident Analysis Methodology (TAC Nos. MA4713, MA4714, and MA4715)”

II.4. Letter from Leonard Olshan (NRC) to Ron Jones (Duke) dated September 24, 2003, “Oconee Nuclear Station, Units 1, 2, and 3 – Safety Evaluation of Revisions to Topical Reports DPC-NE-3000, -3003, and -3005 (TAC Nos. MB5441, MB5442, and MB5443)” (ML032670816)

II.5. Letter from Leonard Olshan (NRC) to Dave Baxter (Duke) dated October 29, 2008, “Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Use of AREVA NP Mark-B-HTP Fuel (TAC Nos. MD7050, MD7051, and MD7052)” (ML082800408)

II.6. Regulatory Guide (RG) 1.24, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure”

II.7. Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors”

II.8. Title 10 Code of Federal Regulations, Section 50.67, “Accident Source Term”

II.9. Letter from NRC (Leonard Olshan) to Duke (Ron Jones) dated June 1, 2004, “Oconee Nuclear Station, Units 1, 2, and 3 RE: Issuance of Amendments (TAC Nos. MB3537, MB3538, and MB3539)” (ML041540097)

II.10. NUREG 0800, Standard Review Plan, Section 6.2.5, Revision 2, “Combustible Gas Control in Containment,” July 1981

II.11. DPC-NE-3000-PA, Revision 5a, “Thermal-Hydraulic Transient Analysis Methodology”, October 2012

II.12. Letter from NRC (Robert Martin) to Duke (H. B. Tucker) dated November 15, 1991, “Safety Evaluation on topical Report DPC-NE-3000, “Thermal-Hydraulic Transient Analysis Methodology” TAC Nos. 73765/73766/73767/73768”

II.13. Letter from NRC (L. A. Wiens) to Duke (M. S. Tuckman) dated August 8, 1994, “Safety Evaluation Regarding the Thermal Hydraulic Transient Analysis Methodology DPC-NE-3000 for Oconee Nuclear Station Units 1, 2, and 3”, TAC Nos. M87112, M87113, and M87114”

II.14. Letter from NRC (Robert Martin) to Duke (M. S. Tuckman) dated December 27, 1995, “Safety Evaluation for Revision 1 to Topical Report DPC-NE-3000-P, “Thermal-Hydraulic

Transient Analysis Methodology” McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station, Units 1 and 2; and Oconee Nuclear Station Units 1, 2, and 3”, TAC Nos. M90143, M90144, and M90145

II.15. Letter from NRC (Dave LeBarge) to Duke (W. R. McCollum) dated October 14, 1998, “Review of Topical Report DPC-NE-3000-PA, Revision 2, “Thermal-Hydraulic Transient Analysis Methodology” – Oconee Nuclear Station, Units 1, 2, and 3”, TAC Nos. MA1127, MA1128, and MA1129

II.16. Letter from Hal B. Tucker (Duke) to Harold R. Denton (NRC) dated December 12, 1984, “Oconee Nuclear Station Docket Nos. 50-269, -270, -287”

II.17. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated June 5, 1985, “NRC Safety Evaluation Report on Duke Response to Generic Letter 81-21 Natural Circulation Cooldown”

II.18. DPC-NE-3003-PA, Rev. 1, “Mass and Energy Release and Containment Response Methodology,” September 2004

II.19. Letter from NRC (L. A. Wiens) to Duke (M. S. Tuckman) dated March 15, 1985, “Safety Evaluation for Topical Report DPC-NE-3003-P, “Mass and Energy Release and Containment Response Methodology” (TAC Nos. M87258, M87259, and M87260)”

II.20. Letter from NRC (Jeffery A. Whited) to Duke (S. Batson) dated April 29, 2016, “Oconee Nuclear Station Units 1, 2, and 3, Issuance of Amendments to Add High Flux Trip for Three Reactor Coolant Pump Operation (CAC Nos. MF6363, MF6364, and MF6365)” (ML16088A330)

II.21. Letter from NRC (John Stang) to Duke (Preston Gillespie) dated July 21, 2011, “Oconee Nuclear Station Units 1, 2, and 3 Re: Issuance of Amendments Regarding Approval for the Use of Gadolinia as an Integral Burnable Absorber (TAC Nos. ME2504, ME2505, and ME2506)” (ML11137A150)

II.22. DPC-NE-2005-PA, Rev. 5, “Thermal-Hydraulic Statistical Core Design Methodology,” March 2016

II.23. BAW-10192P-A, Rev. 0, “BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants”

II.24. Pressurized Water Reactor Owners Group Submittal of BAW-2374-NP, Revision 2 “Risk-Informed Assessment of Once Through Steam Generator Tube Thermal Loads due to Breaks in the Reactor Coolant System Upper Hot Leg Large-Bore Piping” (PA-ASC-0255),” dated January 4, 2007 (ML070330123)

II.25. Letter from Duke (Ron Jones) to NRC dated October 19, 2009, “Oconee Nuclear Site, Units 1, 2, & 3, Docket Nos. 50-269, 50-270, 50-287, Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber, License Amendment Request Number 2009-12”

II.26. Letter from Duke (Preston Gillespie) to NRC dated November 15, 2010, “Oconee Nuclear Site, Units 1, 2, & 3, Docket Nos. 50-269, 50-270, 50-287, Supplement for Proposed License Amendment Request to Revise the Technical Specifications Pursuant to the Use of Gadolinia Integral Burnable Absorber, License Amendment Request Number 2009-12”

II.27. Letter from NRC (David LeBarge) to Duke (R. W. McCollum) dated September 18, 2000, “Oconee Nuclear Station Units 1, 2, and 3 Re: Issuance of Amendments (TAC Nos. MA5348, MA5349, and MA5350)”

II.28. Letter from NRC (Stuart Richards) to EPRI (Gary Vine) dated January 25, 2001, “Safety Evaluation Report on EPRI Topical Report NP-7450(P), Revision 4, “RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems” (TAC No. MA4311)”

II.29. Letter from NRC (David LeBarge) to Duke (R. W. McCollum) dated July 17, 2001, “Oconee Nuclear Station, Units 1, 2, and 3 (ONS) RE: Exemption from the Requirements of Hydrogen Control Requirements of 10 CFR PART 50, Section 10 CFR 50.44, 10 CFR PART 50, Appendix A, General Design Criterion 41, and 10 CFR PART 50, Appendix E Section VI (TAC NOS. MA9635 MA9636, AND MA9637)”

II.30. NUREG/R-2847, "COGAP: A Nuclear Power Plant Containment Hydrogen Control System Evaluation Code," January 1983

II.31. Letter from NRC (C. O. Thomas) to UGRA (T.W. Schnatz) dated September 4, 1984, “Acceptance for Referencing of Licensing Topical Reports EPRI CCM-5, “RETRAN – A Program for One Dimensional Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems” and EPRI NP-1850-CCM, “RETRAN-02 A Program for Transient Thermal-Hydraulic Analysis for Complex Fluid Flow Systems””

II.32. Letter from NRC (A.C. Thadani) to GPU (R. Furia) dated October 19,1988, “Acceptance for Referencing Topical Report EPRI-NP-1850 CCM-A, Revisions 2 and 3 Regarding RETRAN02/MOD003 and MOD004”

II.33. Letter from NRC (A.C. Thadani) to Texas Utilities (James Boatwrite) dated November 1, 1991, “Acceptance for Reference of RETRAN02/MOD005.0”

II.34. Letter from NRC (John Stang) to Duke Energy (Preston Gillespie) dated December 29, 2010, “Oconee Nuclear Station Units 1, 2, and 3, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance With 10 CFR 50.48(c) (TAC Nos. ME3844, ME3845, and ME3846)”

II.35. EDM-102: Instrument Setpoint/Uncertainty Calculations, Revision 3, Engineering Directives Manual

II.36. Letter from NRC (Audrey Klett) to Duke Energy (J. Ed Burchfield Jr), Oconee Nuclear Station, Duke Energy Carolinas, “Oconee Nuclear Station, Units 1, 2, and 3 – Issuance of Amendments Regarding the Updated Final Safety Analysis Report Section for the Standby Shutdown Facility (EPID L-2017-LLA-0365),” dated December 17, 2018 (ML18311A134)

II.37. Duke Energy Letter ONS-2016-081 to the U. S. Nuclear Regulatory Commission from Thomas D. Ray, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, “Completion of Protected Service Water System Milestone 6 and Achievement of Full Compliance with Confirmatory Order EA-13-010,” dated September 21, 2016 (ML16267A448)

II.38. Duke Energy Letter RA-19-0253 to the U. S. Nuclear Regulatory Commission from J. Ed Burchfield Jr, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, “Proposed

License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Breaks Outside of the Containment Building," dated August 28, 2019 (ML19240A814)

II.39. Duke Energy Letter RA-19-0301 to the U. S. Nuclear Regulatory Commission from J. Ed Burchfield Jr., Vice President, Oconee Nuclear Station, Duke Energy Carolinas, "Proposed Amendment to the Renewed Facility Operating Licenses Regarding Revisions to the Updated Final Safety Analysis Report Sections Associated with the Oconee Tornado Licensing Basis – Response to Request for Additional Information," dated July 31, 2019 (ML19217A149)

II.40. Duke Energy Letter RA-18-0026 to the U. S. Nuclear Regulatory Commission from J. Ed Burchfield Jr, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, "Proposed Amendment to the Renewed Facility Operating Licenses Regarding Revisions to the Updated Final Safety Analysis Report Sections Associated with the Oconee Tornado Licensing Basis, License Amendment Request No. 2018-02," dated September 14, 2018 (ML18264A018)

II.41. Duke Letter to the NRC, Subject: Report of Evaluation of the Effects of Steam Line Break on SSF Capabilities, July 9, 1986

II.42. Letter from NRC (Tony Brown) to Duke Energy (Thomas D. Ray), Oconee Nuclear Station, Duke Energy Carolinas, Subject: Oconee Nuclear Station, Units 1, 2, and 3 – Safety Evaluation Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051 (CAC Nos. MF0782, MF0783, MF0784, MF0785, MF0786, and MF0787)," dated August 30, 2017 (ML17202U791)

II.43. Letter from NRC (Audrey Klett) to Duke Energy (J. Ed Burchfield Jr), Oconee Nuclear Station, Duke Energy Carolinas, Subject: Oconee Nuclear Station, Units 1, 2, and 3 – Issuance of Amendments 415, 417, and 416, Regarding the Updated Final Safety Analysis Report Description of Tornado Mitigation (EPID L-2018-LLA-0251)," dated October 31, 2019 (ML19260E084)

II.44. Duke Letter to the NRC, Subject: Supplement to License Amendment Request for Measurement Uncertainty Recapture Power Uprate, License Amendment Request No. 2011-02, Supplement 1, November 21, 2011 (ML11326A296)

II.45. Duke Letter (J. W. Hampton) to the NRC, Subject: Seismic Licensing Basis, dated August 18, 1994 (ML12103A128)

II.46. Letter dated December 5, 1994 from L. A. Wiens (NRC) to J. W. Hampton (Duke Power Company), Subject: Oconee Licensing Basis for Seismic Event and Single Failure Consideration (ML12103A129)

II.47. Letter from Duke to the NRC dated January 31, 2017, Subject: Mitigating Strategies Assessment (MSA) of Oconee Nuclear Station's Flooding Hazard Reevaluation Report (FHRR) (ML17044A016)

II.48. Letter dated July 11, 2017 from Juan Uribe, NRC to T. D. Ray, Duke, Subject: Oconee Nuclear Station, Units 1, 2, and 3 - Flood Hazard Mitigation Strategies Assessment (ML17166A260)

Table II.1-1: Oconee Analyses

UFSAR Section	Analysis Title	Power Used in this Analysis	Is Power Bounding for MUR?	Confirm that bounding event determinations remain valid	Approved by NRC or conducted using methods/processes approved by the NRC	Reference for NRC approval
RIS 2002-03:	**II.1.A**	**II.1.B.i**	**II.1.B.i**	**II.1.C**	**II.1.B.ii**	**II.1.D**
(1) 15.1	Methodology	NA	NA	No bounding event determinations	UFSAR Section discusses analysis methodology, not a specific accident.	Reference II.3
(2) 15.2	Startup Accident	0 MWt	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(3) 15.3	Rod Withdrawal At Power Accident	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(4) 15.4	Moderator Dilution Accidents	Mode 1, Mode 6	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(5) 15.5	Cold Water Accident	2054 MWt (80% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(6a) 15.6	Loss of Coolant Flow Accidents – Flow Coastdown	2054 MWt (80% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4 and II.5
(6b) 15.6	Loss of Coolant Flow Accidents – Locked Rotor	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4 and II.5
(7) 15.7	Control Rod Misalignment Accidents (Dropped Rod)	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(8) 15.8	Turbine Trip Accident	2105.8 MWt (82% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(9) 15.9	Steam Generator Tube Rupture Accident	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(10) 15.10	Waste Gas Tank Rupture Accident	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.6	
(11) 15.11	Fuel Handling Accidents	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	References II.7 and II.8	Reference II.9

UFSAR Section	Analysis Title	Power Used in this Analysis	Is Power Bounding for MUR?	Confirm that bounding event determinations remain valid	Approved by NRC or conducted using methods/processes approved by the NRC	Reference for NRC approval
(12) 15.12	Rod Ejection Accident	0 MWt 1977 MWt (77% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1 References II.7, II.8, II.25, and II.26	References II.2, II.3, II.4, and II.5 Reference II.9
(13) 15.13	Steam Line Break Accident	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, and II.5
(14) 15.14	Loss of Coolant Accidents	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.23	Reference II 23
(15) 15.15	Maximum Hypothetical Accident	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	References II.7 and II.8	Reference II.9
(16) 15.16	Post-Accident Hydrogen Control	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.10	Reference II.29
(17) 15.17	Small Steam Line Break Accident	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.1	References II.2, II.3, II.4, II.5, and II.20
(18) 15.18, 7.8	Anticipated Transients Without Scram	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.11	References II.4, II.5, II.12, II.13, II.14, and II.15
(19) 5.1.2.4	Natural Circulation Cooldown	0 MWt	Yes	See discussion in Section II.1.D.iii	Reference II.16	Reference II.17
(20) 6.2.1.1.3.1 6.2.1.1.3.2 6.2.1.1.3.3	Containment Performance	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.18	References II.19 and II.4
(21)	EQ parameters	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	-See Containment Performance -See NFPA-805 fire -Penetration room – Reference II.11	-See Cont. Performance -See NFPA-805 fire - Pen. Room – References II.4, II.5, II.12, II.13, II.14, and II.15
(22a)	SSF Event Turbine Building Flood (TBF)	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.11	References II.4, II.5, II.12, II.13, II.14, II.15, and II.36

UFSAR Section	Analysis Title	Power Used in this Analysis	Is Power Bounding for MUR?	Confirm that bounding event determinations remain valid	Approved by NRC or conducted using methods/processes approved by the NRC	Reference for NRC approval
(22b)	External Flood	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii		Reference II.42
(23)	Station Blackout (SBO)	See discussion in Section II.1.D.iii	Yes	See discussion in Section II.1.D.iii	See discussion in Section II.1.D.iii	See discussion in Section II.1.D.iii
(24) 9.5.1	NFPA-805 Fire	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.11	References II.4, II.5, II.12, II.13, II.14, and II.15
(25)	Spent Fuel Pool Accidents (loss of pool cooling)	Decay Heat	Yes	See discussion in Section II.1.D.iii	See discussion in Section II.1.D.iii	See discussion above
(26) 10.4.7.3.1	Loss of Main Feedwater	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.11	References II.4, II.5, II.12, II.13, II.14, and II.15
(27) 10.4.7.3., 5.2.3.4	Main Feedwater Line Break	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.18	References II.4 and II.19
(28)	LTOP	0	Yes	See discussion in Section IV.1.C.iv	See Section IV.1.C.iv	
(29) 3.6, 9.6	High Energy Line Break/Pipe Rupture (HELB)	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	References II.11 and II.18	References II.4, II.12, II.13, II.14, II.15, and II.19
(30)	RPS/ESF Instrument Uncertainties	0 MWt	Yes	See discussion in Section II.1.D.iii		
(31)	Natural Phen: Tornado (incl. missiles), Wind, Hurricane	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii	Reference II.18	Reference II.43
(32)	Double Steam Line Break with SSF	2619 MWt (102% of 2568)	Yes	See discussion in Section II.1.D.iii		

III Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level

III.1 **This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding).**

RESPONSE:

See Section II.1.D.iii, Items 1 through 32; and Table II.1-1 items 1 through 32, for discussion of the ONS UFSAR Chapter 15 accident analyses. ONS has no UFSAR Chapter 15 analyses that require re-evaluation for the MUR power uprate.

III.2 **For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:**

III.2.A **Identify the transient/accident that is the subject of the analysis**

III.2.B **Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate**

III.2.C **Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59**

III.2.D **Provide a reference to the NRC's approval of the plant's reload methodology**

RESPONSE:

ONS has no reload analyses that require re-evaluation for the MUR power uprate. Various reload analyses are performed for each fuel cycle in accordance with normal cycle design practice and included in the Core Operating Limits Report per ONS Technical Specification 5.6.5, but there will be no change to those analyses or their methodology based on the MUR power uprate.

III.3 **For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should:**

III.3.A **Identify the transient or accident that is the subject of the analysis**

III.3.B **Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate**

III.3.C **Confirm that the limiting event determination is still valid for the transient or accident being analyzed**

III.3.D **Identify the methodologies used to perform the analyses, and describe any changes in those methodologies**

III.3.E **Provide references to staff approvals of the methodologies in Item D. above**

III.3.F **Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology**

III.3.G **Describe the sequence of events and explicitly identify those that would change as a result of the power uprate**

III.3.H **Describe and justify the chosen single-failure assumption**

III.3.I **Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate**

III.3.J **Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head (NPSH), valve isolation capabilities) required to support the analysis**

III.3.K **Discuss the results and acceptance criteria for the analysis, including any changes from the previous analysis**

RESPONSE:

All ONS calculations of record for the UFSAR Chapter 15 analyses support the MUR power uprate as described in Section II.

IV Mechanical/Structural/Material Component Integrity and Design

IV.1 **A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.**

RESPONSE:

Table IV-1 presents a summary of the primary system critical parameters. MUR power uprate data is shown for maximum analytical thermal power of 2619 MWt (102% of 2568). Licensed thermal power will be approximately 2610 MWt. Summer condenser circulating water conditions and 10% steam generator tube plugging are assumed.

Table IV-1: MUR Power Uprate Critical Parameters

	Current	MUR Power Uprate
Reactor		
Core Thermal Power (MWt)	2568	2619
Other RCS Power (RCP Heat) (MWt)	16	16
Total Thermal Power (MWt)	2584	2635
Reactor Flow (E+06 lb/hr)	145.5	145.52
Reactor Coolant Pressure (psia)	2155	2155
T_{hot} (˚F)	601.7	602.1
T_{cold} (˚F)	556.4	556.0
T_{ave} (˚F)	579.1	579.1
Steam Generators		
Steam Temperature (˚F)	592.0	591.3
Steam Superheat (˚F)	55.64	54.86
Steam Pressure (psia)	925	925
Steam Flow/Feedwater Flow (E+06 lb/hr)	10.91	11.14
Final FW Temperature (˚F)	455.6	458.9

Analyses and evaluations for SSCs performed to support this LAR which were within the scope of the ONS Units 1, 2, and 3 license renewal effort were done in accordance and consistent with, the methodologies approved and referenced in NUREG-1723, Safety Evaluation Report Related to the License Renewal of ONS Units 1, 2, and 3.

IV.1.A This discussion should address the following components:

RESPONSE:

IV.1.A.i Reactor vessel, nozzles, and supports

The revised operating conditions were reviewed for impact on the existing design basis analyses for the reactor vessel. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor vessel remain applicable for the uprated power conditions.

The most recent inspection of the ONS Unit 1 reactor vessel was performed during end of cycle (EOC) 27 refueling outage and the results reported to the NRC (Reference IV.1). The most recent inspection of the ONS Unit 2 reactor vessel was performed during EOC 27 refueling outage and the results reported to the NRC (Reference IV.2). The most recent inspection of the ONS Unit 3 reactor vessel was performed during EOC 27 refueling outage and the results reported to the NRC (Reference IV.3).

IV.1.A.ii Reactor core support structures and vessel internals

Reactor internal components include the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, and thermal shield.

Operating T_{ave} (coolant temperature in the center of the core) remains unchanged while there is a slight increase in operating T_{hot} (602.1°F core exit temperature) and a slight decrease in operating T_{cold} (556.0°F core inlet temperature). The core delta temperature will experience a nominal operating increase in order to remove the MUR power increase, but the revised core parameters are bounded by the design values plus uncertainty that were used in the current analyses. Therefore, the reactor vessel internals operation after the MUR power increase is bounded by the current normal operation analyses.

The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the reactor vessel internals. No changes to the RCS pressure were made as part of the power uprate. The existing analyses are based on the design conditions in the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid.

The structural adequacy of RV internals and incore instrument nozzles of ONS Units 1, 2 and 3 was also reviewed with respect to flow induced vibration (FIV) relative to the MUR power uprate. The components currently analyzed for FIV include the incore instrumentation nozzles, the flow distributor assembly, the thermal shield, and the inlet baffle. From the comparative analysis, the

new operational condition of ONS Units 1, 2 and 3 after the MUR power uprate are bounded by the current analysis (topical report BAW-10051). The RV internals and incore instrument nozzles are structurally adequate with regard to flow-induced vibration including the effects of the MUR power uprate.

MRP-227-A (Reference IV.3) documents plant-specific implemented requirements imposed by the industry under NEI 03-08 (Reference IV.5) to manage reactor vessel internals aging. The ONS RVI Inspection Plan was approved by the NRC in an SER dated 6/19/2015 (Reference IV.4). This Inspection Plan is described in UFSAR Section 18.3.20. A review of the RVI Inspection Plan concluded that it was not affected by the MUR power increase.

IV.1.A.iii Control rod drive mechanisms

The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the control rod drive mechanisms. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot}/T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the control rod drive mechanism remain applicable for the uprated power conditions.

IV.1.A.iv Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles

The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the reactor coolant piping and supports. No changes in RCS design or operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot} and T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor coolant piping and supports remain applicable for the uprated power conditions.

There is a discussion of thermal stratification and Bulletin 88-11, in Section IV.1.B.iv.

IV.1.A.v Balance-of-plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, and containment systems)

The structural analyses of the piping attached to the RCS (decay heat line, makeup and purification line, high and low pressure injection lines) use anchor motions from the RCS structural analyses. These anchor motions do not change due to the MUR power uprate power conditions. The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant system attached piping and supports. No changes to the RCS design or operating pressure were made as part of the MUR power uprate. The effects of the operation temperature changes (T_{hot} and T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate

conditions are bounded by the design conditions. Since the operation transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid.

The Chemical Addition (CA) System will continue to perform its safety related functions of containment isolation and post-accident sump pH control after the power uprate. The containment analyses presented in UFSAR Section 6.2 were performed at 102% of rated power, bounding the power uprate for pressure and temperature loads on the containment isolation valves. The sizing of the tri-sodium phosphate baskets remains valid at 102% of rated power.

The Core Flood (CF) System will continue to perform its safety function of emergency core cooling. The core flood tanks contents are injected into the RCS following several postulated design bases events. These event analyses assumed a power level of 2% above the licensed power for peak containment pressure mass and energy releases. Therefore, the CF System function during these events is bounded by existing analyses for the MUR. There is no impact to this system due to the MUR.

The design bases of the Coolant Storage (CS) System will remain valid after the MUR. The reactor coolant bleed holdup tank sizing and the quench tank sizing are not impacted by the increased power level. The CS System will continue to be able to perform its safety function of containment isolation after the MUR power uprate.

The High Pressure Injection (HPI) System will continue to perform its core cooling and shutdown functions in response to specified design basis accidents. The HPI System reactor coolant pressure boundary and containment isolation barrier functions will not be impacted by the MUR. The design bases of the HPI System identified in UFSAR Section 6.3 and 9.3.2 will remain valid after the MUR. The operational letdown, makeup, and purification functions of the HPI System are not impacted by the MUR.

The Low Pressure Injection (LPI) System will continue to perform its safety functions in response to specified design basis accidents. The system will also maintain its ability to provide decay heat removal during plant shutdown. The margin between the design and operating heat loads is sufficient to allow for the increase to the MUR power level.

The Post Accident Sampling (PAS) System will continue to perform its safety related functions of maintaining a containment isolation boundary during normal and upset conditions after the MUR power uprate. The source of the PAS samples, RCS, operating flow, temperature, and pressure prior to and following a DBE are unchanged. There is no adverse impact to this system due to the MUR power uprate.

The RCS will continue to operate in order to perform its safety related function of maintaining a fission product boundary during normal and upset conditions after the MUR power uprate. The RCS operating flow, temperature, and pressure prior to and following a DBE are unchanged.

There is no impact to the design basis or operation of this system due to the MUR power uprate.

The SSF Reactor Coolant Makeup (RCM) System described in UFSAR Section 9.6 will continue to perform its safety related functions of RCP seal injection and replenishing the RCS inventory

after the MUR power uprate. The existing RCS analyses were performed at 2619 MWt (102% of the original core thermal power of 2568 MWt); therefore, flow, temperature, and pressure experienced by the SSF RCM System and its components will be bounded for a MUR power uprate. There is no adverse impact to this system due to the MUR power uprate.

Containment Systems are discussed in Section VI.1.B.

Safety-related cooling water systems are discussed in Section VI.1.C.

IV.1.A.vi Steam generator tubes, secondary side internal support structures, shell, and nozzles

The MUR conditions are bounded by the thermal hydraulic conditions used as the design basis for the Replacement Once-Through Steam Generators (ROTSGs). No new transients have been proposed for the MUR and ramp rates have not been increased.

Therefore, the existing design basis remains bounding for MUR conditions.

The ROTSG structural analyses of pressure boundary components (including the tube flaw size analysis required by Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, and the analyses of tube plugs and tube stabilizers) meet the requirements of the existing design basis and are therefore bounding for MUR conditions. The tube-to-shell interaction analysis meets the existing design basis and is therefore bounding for MUR conditions. Existing loads remain valid, and stresses and fatigue values for all pressure boundary components remain valid and applicable for MUR conditions. Furthermore, existing tube loads for faulted conditions including LOCA and MSLB accident conditions are bounding for MUR conditions. The ROTSG pressure boundary was certified in accordance with ASME Section III, Division 1, 1989 Edition, no Addenda. This ASME Code edition remains applicable to the ROTSG components.

The ROTSG structural analyses of internal components meet the requirements of the existing design basis and are therefore bounding for MUR conditions. Existing loads remain valid, and stresses and fatigue values for all internal components remain valid and applicable for MUR conditions.

The ROTSG seismic analysis meets the requirements of the existing design basis and is therefore bounding for MUR conditions.

The thermal hydraulic analyses meet the requirements of the existing design basis and are therefore bounding for MUR conditions. The flow induced vibration (FIV) analyses meet the requirements of the existing design basis and are therefore bounding for MUR conditions. See Section IV.1.F below for further discussion of Flow Induced Vibration and Tube Wear.

IV.1.A.vii Reactor coolant pumps

The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the reactor coolant pump. No changes to the RCS operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot} and T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS

functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Therefore, the existing stress reports for the reactor coolant pump remain applicable for the power uprate conditions.

The MUR power uprate doesn't result in a measurable reactor coolant mass flow change (~ 0.014%), but the amount of work expended by the RCPs theoretically increases due to the increased density of the coolant at T_{cold} conditions. However, this increase in pump work, as demonstrated on other MUR power uprates, will not be measurable with regards to pump motor current and therefore there is no adverse impact on the RCPs. The pump seals will continue to operate under the same normal operating conditions and will subsequently see no change in seal leakage or susceptibility to seal failures during normal or DBE conditions.

IV.1.A.viii Pressurizer shell, nozzles, and surge line

The MUR power uprate conditions were reviewed for impact on the existing design basis analyses for the pressurizer. No changes to the RCS operating pressure were made as part of the power uprate. The effects of operating temperature changes (T_{hot} and T_{cold}) are within design limits. The design conditions in the existing analyses are based on the RCS functional specification. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Therefore, the existing stress reports for the pressurizer remain applicable for the power uprate conditions.

The RCS includes the pressurizer and associated pressure relief valves that protect the reactor vessel by limiting the post-accident vessel pressure below the design limits of the RCS pressure boundary. The pressurizer is designed to maintain the RCS pressure and accommodate the shrink and swell of the RCS that occurs following a reactor trip. The pressurizer controls the RCS pressure by maintaining the temperature of the pressurizer liquid at the saturation temperature corresponding to the desired system pressure. Pressurizer temperature control is maintained by heaters and spray. The pressurizer heaters supply energy to heat the pressurizer liquid to the required temperature. The pressurizer spray functions to cool the pressurizer by injecting water from the RCS cold leg into the steam space if the pressurizer pressure should increase above its desired pressure during transients. Small reactor coolant pressure and volume compensations are made by providing steam volume to absorb flows into the pressurizer and water volume to match flows out of the pressurizer.

The impact on the full power RCS mass and the pressurizer spray requirements were evaluated due to the MUR power increase and the impact on T_{cold}. The slight decrease in the full power operation T_{cold} (~ 0.4°F) is conservative with regards to pressurizer spray valve performance (cooler spray water is more effective) therefore there is no adverse impact on pressurizer pressure control functions due to the MUR power increase. The reduction in pressurizer spray temperature will not cause additional pressurizer fatigue cycles or otherwise shorten the pressurizer vessel life since the temperature is within the normal operating band.

The pressurizer will continue to perform its safety related function after the MUR power uprate.

The RCS operating pressure is unchanged. There is no adverse impact to the pressurizer due to the MUR power uprate.

There is a discussion of thermal stratification and Bulletin 88-11 in Section IV.1.B.iv.

IV.1.A.ix Safety-related valves

The pressurizer code safety valves, power operated valves, and block valves located on top of the pressurizer provide over pressure protection for the RCS. The changes due to the MUR power increase that could potentially impact the pressurizer valves are RCS mass and reactor power (including RCP heat). The RCS mass does not significantly change due to the MUR power increase, based on the small changes in T_{hot} and T_{cold}. The MUR power uprate is bounded by the current design basis event transient analyses (Section II), and thus there is no adverse impact on the pressurizer overpressure protection valves from the MUR power uprate. Based on this review, it was determined that the analysis of record for the pressurizer overpressure protection valves remains bounding at MUR power uprate conditions.

Other safety-related valves were reviewed as part of the system that contains those valves. As discussed in Sections IV.1.A.v and VI.1, operating conditions for interfacing systems will see small to no change under MUR power uprate conditions. Based on these reviews, it was determined that the analysis of record for interfacing system valves remain bounded at MUR conditions.

IV.1.B The discussion should identify and evaluate any changes related to the power uprate in the following areas:

RESPONSE:

IV.1.B.i Stresses

No changes in the RCS design or operating transient conditions were made as part of the MUR power uprate. The design condition analyses are based upon the RCS functional specification. Considering the margins for the primary, primary plus secondary stresses and fatigue usage factors in the RCS piping loop, it is concluded that the MUR power uprate conditions remain acceptable and are bounded by the design conditions.

IV.1.B.ii Cumulative usage factors

The revised design conditions for the NSSS components, piping and interface systems were reviewed for impact on the existing design basis analyses. For NSSS components, the evaluation showed that the operating conditions due to the MUR power uprate are bounded by those used in the existing analyses. Further, since the evaluated transients will not change as a result of the power uprate, the existing loads remain valid, and the stresses and fatigue values (cumulative usage factors) also remain valid.

There is a discussion of thermal stratification and Bulletin 88-11 in Section IV.1.B.iv.

IV.1.B.iii Flow induced vibration (FIV)

For the increase in power from 2568 MWt to 2619 MWt (which bounds MUR power uprate conditions), the RCS mass flow rate did not increase for the "0% Tube Plugging" condition and increased by 0.014% for the "10% Tube Plugging" condition. The 0.014% increase in the mass flow rate is insignificant.

Currently, flow induced vibration concerns are limited to the reactor vessel internals and the steam generator tubes. FIV of the reactor vessel internals and incore instrument nozzles is discussed in Section IV.1.A.ii. FIV of the once-through steam generators is discussed in Section IV.1.F consistent with the RIS 2002-03 outline.

IV.1.B.iv Changes in temperature (pre- and post-uprate)

Temperature Changes:

The changes in operating temperatures are provided in Table IV-1. The average temperature is unchanged, and the cold leg decreases 0.4 °F, while the hot leg temperature increases 0.4 °F. These changes, as discussed elsewhere, have minimal impact on the MUR power uprate.

Evaluation of Potential for Thermal Stratification:

Thermal stratification in the lines attached to the primary side of the RCS occurs mainly during heatup and cooldown. The current 100% power hot and cold leg operating temperatures that the plant has been designed to are essentially the same as those for the MUR power uprate. This means that the effects of thermal stratification will not change as a result of the power uprate.

NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems”, addresses the issue of thermal stresses in piping attached to the primary loop due to turbulent penetration. The temperature changes as a result of the MUR power uprate compared to the current operation are negligible and will not have an adverse effect on the existing or potential thermal stress or stratification conditions. In addition, the design RCS flow rates are unchanged for the MUR power uprate. Therefore, the effects of the turbulent penetration will not change as a result of power uprate.

NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification”, addresses the issue of surge line thermal stratification. Thermal stratification in the surge line occurs mainly during plant heatup and cooldown and is driven by the temperature difference between the hot leg and the pressurizer. The current operating temperature of the hot leg will increase very slightly due to MUR power uprate. A higher hot leg temperature gives a lower temperature differential between the hot leg and the pressurizer, which in turn lessens the stratification effects. This means that stress and fatigue in the surge line which is attributed to thermal stratification is bounded by the existing analyses.

The RCP motor load increases slightly from reduced cold leg temperature (increased density). The slight increase in RCP motor load is bounded by the current operating analysis.

IV.1.B.v Changes in pressure (pre- and post-uprate)

The system operating pressures remain unchanged as shown in Table IV-1.

IV.1.B.vi Changes in flow rates (pre- and post-uprate)

As provided in Table IV-1, there is no change in design RCS flow for the MUR power uprate. The change in nominal full power flow is less than 0.1%. This small change in mass flow rate will have no impact on core design and safety analyses. A detailed review of safety analyses is provided in Section II.

IV.1.B.vii High energy line break (HELB) locations

The impact of the MUR power uprate on HELB locations inside and outside containment at ONS was evaluated. As discussed in Sections IV.1.B.i and IV.1.B.ii, the design conditions are not changing for the MUR power uprate, and thus the stresses on components and the cumulative usage factors for those components are not changing. Thus, there is no impact on HELB locations. The results of this HELB evaluation are presented in Section II of this enclosure.

The MUR power uprate is bounded by the existing HELB analysis of record for ONS Units 1, 2, and 3 as discussed in Section II.

IV.1.B.viii Jet impingement and thrust forces

The Leak-Before-Break (LBB) concept applies known mechanisms for flaw growth to piping designs with assumed through-wall flaws and is based on the plants ability to detect an RCS leak. Topical report BAW-1847 Rev. 1 presents the LBB evaluation of the RCS primary piping. It showed that a double-ended guillotine break will not occur and that postulated flaws producing detectable leakage exhibit stable growth, and thus, allow a controlled plant shutdown before any potential exists for catastrophic piping failure. The major areas that contributed to the evaluation are the RCS piping structural loads, leakage flaw size determination, material properties, and flaw stability analysis. An evaluation was performed which determined that the impact of the MUR power uprate design conditions on the inputs to the LBB analyses is negligible, and the LBB conclusions remain unchanged.

RCS components were previously acceptable for Loss-of-Coolant Accident (LOCA) loadings including the T_{ave} reduction and 102% power conditions. Due to the LBB qualification, the breaks considered were limited break ruptures of the smaller attached piping (core flood, decay heat, surge line, steam line, and feedwater line). The MUR power uprate design conditions were reviewed for impact on the existing hydraulic forcing functions. It was determined that the existing temperatures are more controlling and the loads remain bounded by the values in the existing analyses.

IV.1.C **The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:**

RESPONSE:

Note: On 2/27/2014, the NRC issued an SE (Reference IV.9) that revised the ONS P-T limit curves to 54 EFPY that corresponded to ONS's extended 60-year licenses. The CLB for RT PTS remained at 48 EFPY. As discussed below, some reactor vessel material evaluations for the MUR power uprate were performed at 72 EFPY. The CLB for the specific evaluation is not extended to 72 EFPY. The acceptability of the analysis at 72 EFPY is used to bound analyses at lower EFPY levels.

IV.1.C.i Pressurized thermal shock calculations

The CLB pressurized thermal shock reference temperature (RT_{PTS}) for ONS Units 1, 2, and 3 reactor vessel materials was performed for 48 EFPY and is included in the licensing renewal application (Reference IV.7). At 48 EFPY, all reactor vessel materials meet the 10CFR50.61 screening criteria. To address the impact of MUR, the projected 72 EFPY (80 calendar years) fluences with MUR power uprate were used to calculate RT_{PTS} per 10 CFR 50.61, which resulted in all reactor vessel materials meeting the 10 CFR 50.61 screening criteria, as shown in Tables IV.1-2, IV.1-3, and IV.1-4. Note that all reactor vessel materials with a wetted surface fluence $\geq$ 1.0E+17 n/cm^2 at 72 EFPY with MUR power uprate were considered. Therefore, because the 48 EFPY fluences with MUR power uprate are less than the 72 EFPY fluences with MUR, it is concluded that the ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline materials also meet the 10 CFR 50.61 screening criteria through 48 EFPY with MUR, i.e., the fluences at 72 EFPY with MUR bound those at 48 EFPY with MUR power uprate. Note that, while 72 EFPY with MUR fluences were used in the RT_{PTS} calculations, this submittal does not extend the applicability of these analyses beyond the CLB duration of 48 EFPY for RT_{PTS}.

Based on this analysis, the applicability for the ONS Units 1, 2, and 3 RT_{PTS} calculations considering MUR are as follows:

ONS Unit 1: 48 EFPY

ONS Unit 2: 48 EFPY

ONS Unit 3: 48 EFPY

Table IV.1-2: ONS-1 Predicted RT_{PTS} using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm^2 (E > 1.0 MeV) 72 EFPY with MUR	RT_{PTS} (˚F)
Lower Nozzle Belt (LNB) Forging	2.68E+18	<270
ONFs	3.49E+17	<270
INFs	1.62E+17	<270
IS Plates	1.85E+19	<270
Upper Shell (US) Plates	2.10E+19	<270
Lower Shell (LS) Plates	2.10E+19	<270
Dutchman Forging	2.70E+17	<270
LNB to ONF Welds	3.49E+17	<270
LNB to INF Welds	1.62E+17	<270
LNB to IS Circ. Weld	2.91E+18	<300
IS Longit. Welds	1.38E+19	<270
IS to US Circ. Weld (ID 61%)	1.86E+19	<300
IS to US Circ. Weld (OD 39%)	1.86E+19	<300
US Longit. Welds	1.36E+19	<270
US to LS Circ. Weld	2.05E+19	<300
LS Longit. Welds	1.68E+19	<270
LS to Dutchman Circ. Weld	2.70E+17	<300

Table IV.1-3: ONS-2 Predicted RT_{PTS} using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm^2 (E > 1.0 MeV) 72 EFPY with MUR	RT_{PTS} (˚F)
LNB Forging	1.74E+19	<270
US Forging	1.98E+19	<270
LS Forging	1.97E+19	<270
ONF	3.24E+17	<270
INF	1.50E+17	<270
Dutchman Forging	2.50E+17	<270
LNB to ONF Welds	3.24E+17	<270
LNB to INF Welds	1.50E+17	<270
LNB to US Circ. Weld (100%)	1.75E+19	<300
US to LS Circ. Weld (100%)	1.92E+19	<300
LS to Dutchman Circ. Weld (100%)	2.50E+17	<300

Table IV.1-4: ONS-3 Predicted RT_{PTS} using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm^2 (E > 1.0 MeV) 72 EFPY with MUR	RT_{PTS} (˚F)
LNB Forging	1.81E+19	<270
US Forging	2.06E+19	<270
LS Forging	2.05E+19	<270
ONF	3.50E+17	<270
INF	1.62E+17	<270
Dutchman Forging	2.68E+17	<270
LNB to ONF Welds	3.50E+17	<270
LNB to INF Welds	1.62E+17	<270
LNB to US Circ. Weld (100%)	1.82E+19	<300
US to LS Circ. Weld (ID 75%)	2.01E+19	<300
US to LS Circ. Weld (OD 25%)	2.01E+19	<300
LS to Dutchman Circ. Weld (100%)	2.68E+17	<300

IV.1.C.ii Fluence evaluation

The fluence analysis methodology from BAW-2241P-A (Reference IV.8) was used to calculate the fast neutron fluence (E > 1.0 MeV) of the reactor vessel welds and forgings of interest. The fast neutron fluence at each location was calculated in accordance with the requirements of U.S. Nuclear Regulatory Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Fluence results were calculated for ONS Unit 1 cycles 27-29, ONS Unit 2 cycles 25-28, and ONS Unit 3 cycles 26-28 using a computer model that extends from below the core to the vessel mating surface. These data were benchmarked against cavity dosimetry data from ONS Unit 2 Cycles 25-28. To extrapolate the fluence values to end of life, the last cycle design for each unit was utilized to develop flux values at each location. These flux values were used to extrapolate the fluence to 54 and 72 EFPY assuming 100% power at 2,612 MWt and a partial low leakage core design whereby High Thermal Performance fuel assemblies were on the periphery.

IV.1.C.iii Heatup and cooldown pressure-temperature limit curves

In accordance with 10 CFR 50 Appendix G, P-T limits are based on RT_{NDT} values that are adjusted for the effect of neutron irradiation. The CLB P-T limits for ONS Units 1, 2, and 3 are for 54 EFPY (References IV.9 and IV.10). The limiting ART values of the ONS Units 1, 2, and 3 reactor vessel materials used in the CLB P-T limits are identified in Table IV.1-5 (References IV.10 and IV.11).

Table IV.1-5: Limiting Adjusted Reference Temperatures for the CLB P-T Limits

Vessel Component	Wall Location	Limiting ART (°F) for Each RV Postulated Flaw Category		
		ONS Unit 1	ONS Unit 2	ONS Unit 3
Beltline Axial Weld/Base Metal	1/4T	171.0 (Weld SA-1493)	161.8 (Base Metal AMX-77)	190.8 (Base Metal 4680)
	3/4T	132.9 (Base Metal C2197-2)	135.7 (Base Metal AMX-77)	160.0 (Base Metal 4680)
Beltline Circ. Weld	1/4T	164.2 (Weld SA-1229)	193.1 (Weld WF-25)	195.6 (Weld WF-67)
	3/4T	132.1 (Weld WF-25)	132.5 (Weld WF-25)	162.1 (Weld WF-70)
Nozzle Belt Upper (t=12-inches)	1/4T	111.9 (Base Metal AHR-54)	102.4 (Base Metal AMX-77)	106.3 (Base Metal 4680)
	3/4T	83.5 (Base Metal AHR-54)	79.4 (Base Metal AMX-77)	88.8 (Base Metal 4680)
Outlet/Inlet Nozzle	Wetted Surface	60	60	60

The updated 54 EFPY fluence values with MUR Power Uprate for numerous reactor vessel materials are greater than the fluences used to support the current licensing bases P-T limits. Note that all reactor vessel materials with a wetted surface fluence $\geq$ 1.0E+17 n/cm^2 at 54 EFPY with MUR power uprate were considered. To maintain the validity of the existing limiting ART values and P-T analyses (i.e., current licensing basis), the EFPY applicability was reduced to a time at which the limiting ART values (shown in Table IV.1-5), and thus the existing P-T curves, are maintained.

Based on the updated 54 EFPY fluence with MUR power uprate calculations and the associated fluence rate, the updated applicability for the ONS Units 1, 2, and 3 P-T limit curves are as follows:

ONS Unit 1: 44.6 EFPY

ONS Unit 2: 45.3 EFPY

ONS Unit 3: 43.8 EFPY

The approximate 10 EFPY reduction in the applicability of the P-T limit curves is not due solely to the small increase in fluence from MUR, but is instead primarily due to consideration of NRC Regulatory Issue Summary (RIS) 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components”

(Reference IV-17). RIS 2014-11 states that "the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1.0E+17 n/cm^2 (E > 1.0 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period." The CLB P-T limits were approved prior to issuance of RIS 2014-11 and, therefore, only considered the reduction in toughness due to irradiation for reactor vessel materials located adjacent to the core, which did not include the reactor vessel nozzles. Therefore, to maintain the CLB P-T curves, the EFPY applicability was reduced until the reactor vessel nozzle fluence reached 1.0E+17 n/cm^2 (E > 1.0 MeV).

IV.1.C.iv Low Temperature Overpressure Protection

As described in Section IV.1.C.iii above, the current low-temperature overpressure protection (LTOP) limits in the existing P-T curves do not need to be modified for the updated 54 EFPY fluence with MUR beyond the noted reduction in EFPY applicability.

IV.1.C.v Upper Shelf Energy

The CLB for upper shelf energy (CvUSE) for ONS Units 1, 2, and 3 reactor vessel materials was performed for 48 EFPY in the licensing renewal application (Reference IV.7) and then extended to 54 EFPY (References IV.9 and IV.10). The predicted CvUSE at the 1/4-thickness (1/4T) wall locations for ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline materials were calculated using the guidance in Position 1.2 and Position 2.2 of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," and the projected 72 EFPY (80 calendar years) fluences with MUR, as shown in, Table IV.1-7, and Table IV.1-8. Note that all reactor vessel materials with a wetted surface fluence $\geq$ 1.0E+17 n/cm^2 at 72 EFPY with MUR were considered. Also note that, while 72 EFPY with MUR power uprate fluences were used in the CvUSE calculations, this justification does not extend the applicability of these analyses beyond the CLB duration of 54 EFPY for CvUSE.

Table IV.1-6: ONS-1 Predicted CvUSE using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm2 (E > 1.0 MeV) 72 EFPY with MUR	Is CvUSE > 50 ft-lbs?
LNB Forging	2.68E+18	Yes
ONFs	3.49E+17	Yes
INFs	1.62E+17	Yes
IS Plates	1.85E+19	Yes
US Plates	2.10E+19	Yes
LS Plates	2.10E+19	Yes
Dutchman Forging	2.70E+17	Yes
LNB to ONF Welds	3.49E+17	EMA
LNB to INF Welds	1.62E+17	EMA
LNB to IS Circ. Weld	2.91E+18	EMA
IS Longit. Welds	1.38E+19	EMA
IS to US Circ. Weld (ID 61%)	1.86E+19	EMA
IS to US Circ. Weld (OD 39%)	1.86E+19	EMA
US Longit. Welds	1.36E+19	EMA
US to LS Circ. Weld	2.05E+19	EMA
LS Longit. Welds	1.68E+19	EMA
LS to Dutchman Circ. Weld	2.70E+17	EMA

Table IV.1-7: ONS-2 Predicted CvUSE using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm2 (E > 1.0 MeV) 72 EFPY with MUR	Is CvUSE > 50 ft-lbs?
LNB Forging	1.74E+19	Yes
US Forging	1.98E+19	Yes
LS Forging	1.97E+19	Yes
ONFs	3.24E+17	Yes
INFs	1.50E+17	Yes
Dutchman Forging	2.50E+17	Yes
LNB to ONF Welds	3.24E+17	EMA
LNB to INF Welds	1.50E+17	EMA
LNB to US Circ. Weld (100%)	1.75E+19	EMA
US to LS Circ. Weld (100%)	1.92E+19	EMA
LS to Dutchman Circ. Weld (100%)	2.50E+17	EMA

Table IV.1-8: ONS-3 Predicted CvUSE using 72 EFPY with MUR Power Uprate Fluence

Location	Wetted Surface Fluence, n/cm2 (E > 1.0 MeV) 72 EFPY with MUR	Is CvUSE > 50 ft-lbs?
LNB Forging	1.81E+19	Yes
US Forging	2.06E+19	Yes
LS Forging	2.05E+19	Yes
ONFs	3.50E+17	Note a
INFs	1.62E+17	Yes
Dutchman Forging	2.68E+17	Note a
LNB to ONF Welds	3.50E+17	EMA
LNB to INF Welds	1.62E+17	EMA
LNB to US Circ. Weld (100%)	1.82E+19	EMA
US to LS Circ. Weld (ID 75%)	2.01E+19	EMA
US to LS Circ. Weld (OD 25%)	2.01E+19	EMA
LS to Dutchman Circ. Weld (100%)	2.68E+17	EMA

Note a: The EFPY applicability of the ONF and Dutchman forging are discussed below.

Since the predicted CvUSE at 72 EFPY with MUR power uprate are all above 50 ft-lbs for the ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline base metal materials (except for the ONS Unit 3 outlet nozzle forgings and the Dutchman forging), it is concluded that the predicted CvUSE through 54 EFPY with MUR power uprate would also be above 50 ft-lbs, i.e., the fluences at 72 EFPY with MUR bound those at 54 EFPY with MUR.

For the ONS Unit 3 Dutchman forging, the updated 54 EFPY fluence with MUR of 1.33E+17 n/cm^2 was used to confirm that the predicted CvUSE remains above 50 ft-lbs at 54 EFPY with MUR.

For the ONS Unit 3 outlet nozzle forgings, the CLB fluence was considered less than 1.0E+17 n/cm^2 and, therefore, no embrittlement assessment was required. A corner flaw is considered the limiting postulated flaw in the outlet nozzle. Based on this location, the projected EFPY to reach 1.0E+17 n/cm^2 with MUR power uprate for a postulated corner flaw in the ONS Unit 3 outlet nozzle is 43.8 EFPY.

As for the ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline weld metals, the predicted CvUSE decreases are expected to be below 50 ft-lbs at 54 EFPY with MUR power uprate. Therefore, an equivalent margin analysis (EMA) was performed to demonstrate that the lower values will provide adequate margins of safety.

The equivalent margins analysis for the ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline Linde 80 weld metals using projected 72 EFPY (80 calendar years) fluences are reported in References IV.612 and IV.13. The results of these assessments show all the Linde 80 weld metals in the ONS Units 1, 2 and 3 reactor vessel beltline and extended beltline region provide adequate margins of safety. Therefore, because the 54 EFPY fluences with

MUR are less than the 72 EFPY fluences, it is concluded that the ONS Units 1, 2, and 3 reactor vessel beltline and extended beltline Linde 80 weld metals provide adequate margins of safety through 54 EFPY with MUR power uprate, i.e., the fluences at 72 EFPY bound those at 54 EFPY with MUR power uprate. Note that, while 72 EFPY fluences were used in the EMA calculations, this submittal does not extend the applicability of these analyses beyond the CLB duration of 54 EFPY for EMA.

Based on this analysis, the applicability for the ONS Units 1, 2, and 3 CvUSE/EMA calculations considering MUR are as follows:

ONS Unit 1: 54 EFPY

ONS Unit 2: 54 EFPY

ONS Unit 3: 43.8 EFPY

IV.1.C.vi Surveillance capsule withdrawal schedule

ONS Units 1, 2, and 3 participate in the Pressurized Water Reactor Owners Group (PWROG) Master Integrated Reactor Vessel Surveillance Program (MIRVP) (References IV-14 and IV-15). The NRC has concluded that the MIRVP met the criteria provided by Appendix H to 10 CFR Part 50 and that the current surveillance capsule withdrawal schedule in Reference IV.615 for ONS Units 1, 2, and 3 satisfies the ASTM Standard E185-82 (Reference IV.6).

In accordance with the current capsule withdraw schedule, a “fifth” surveillance capsule supports the extended operation (i.e., 60-year operation) for ONS Units 1, 2, and 3 (Reference IV.15). These surveillance capsules and their respective fluences for the limiting weld data needed are as follows:

ONS Unit 1: MIRVP Capsule CR3-LG2	1.67E+19 n/cm^2 (E > 1.0 MeV), Weld ID: SA-1585
ONS Unit 2: MIRVP Capsule A5	2.75E+19 n/cm^2 (E > 1.0 MeV), Weld ID: WF-25
ONS Unit 3: MIRVP Capsule CR3-LG2	1.95E+19 n/cm^2 (E > 1.0 MeV), Weld ID: WF-67

Per the updated fluence analysis, the 54 EFPY peak fast neutron fluences (E > 1.0 MeV) with MUR for the ONS Units 1, 2, and 3 reactor vessels are:

ONS Unit 1: <1.67E+19 n/cm^2 (E > 1.0 MeV)

ONS Unit 2: <2.75E+19 n/cm^2 (E > 1.0 MeV)

ONS Unit 3: <1.95E+19 n/cm^2 (E > 1.0 MeV)

Comparison of the “fifth” surveillance capsule to the updated 54 EFPY peak fast neutron fluences with MUR power uprate for the ONS Units 1, 2, and 3 reactor vessels shows that these capsules continue to meet the ASTM E185-82 criterion, which states that the fifth capsule may be removed when the capsule neutron fluence is between one and two times the limiting fluence calculated for the vessel at end of life.

Based on this analysis, the applicability for the ONS Units 1, 2, and 3 reactor vessel surveillance program considering MUR power uprate are as follows:

ONS Unit 1: 54 EFPY

ONS Unit 2: 54 EFPY

ONS Unit 3: 54 EFPY

IV.1.C.vii Under clad cracking (not listed in RIS 2002-03)

The impact of the MUR power uprate on the previous reactor vessel underclad cracking (UCC) analysis performed for 48 EFPY without the MUR power uprate was assessed. The underclad cracking analysis of record was performed for B&W-designed plants including the ONS units. The underclad cracking analysis was performed for two regions of the reactor vessel: the beltline region and the nozzle belt region. Since the closure head for each ONS unit has been replaced, and the replacement closure head clad welding is not susceptible to UCC, the closure heads were not re-evaluated. For each of the two regions it was determined that the limiting adjusted reference temperature (ART) at the inside surface of the ONS reactor vessels is less than that used in the previous generic analysis. At 48 EFPY with MUR, the limiting ART value for the nozzle belt region is 172.6 °F, whereas the ART value used in the previous analysis was 175 °F.

The applied stress intensity factors of the postulated underclad cracks in the RV are determined by the RV stresses due to: (a) pressure and thermal transients, (b) any geometric discontinuities, and (c) piping (or nozzle) loads. The transient stresses and the discontinuity stresses have not changed since the original UCC evaluation. However, the stresses due to piping (or nozzle) loads in the RV have changed with the steam generator replacements. The external loads/stresses induced in the RV shell with Steam Generator Replacement and the original external loads/stresses are compared against previous results and ASME Code requirements. The results demonstrate that the conclusions of the previous UCC evaluation remain valid for the MUR power uprate. The results also demonstrate that the postulated flaws in the reactor vessel meet the acceptance criteria of ASME Code Section XI, paragraph IWB-3612 (Reference IV-18). The fracture toughness margin for normal and upset condition was determined to be 3.63, which is greater than the required fracture toughness margin of $\sqrt{10}$. The fracture toughness margin for the emergency and faulted condition was determined to be 1.55, which is greater than the required fracture toughness margin of $\sqrt{2}$. Hence, the postulated underclad cracks in the reactor vessel for all three Oconee Units have been found to be acceptable by the IWB-3612 criteria for continued safe operation following MUR power uprate.

_IV.1.D_ The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

No stress/fatigue analyses were revised, and hence no code of record changed. The codes of record remain as stated in UFSAR Table 5-4, Reactor Coolant System Component Codes.

Table IV.1.D-1: Codes of Record

Component	Code	Edition and Addenda
Reactor Vessel	ASME III Class A	Summer 1967[1]
Replacement Reactor Vessel Head	ASME III Class 1	1989, No addendum[3,4]
Pressurizer	ASME III Class A	Summer 1967[1]
Reactor Coolant System Piping	USAS B31.7	Errata through June 1968[5]
Feedwater Header	USAS B31.1	1967
R. C. Pump Casings	ASME III Class A (not code stamped)	Summer 1967
Safety and Relief Valves	ASME III Art. 9	Summer 1967
Welding Qualifications	ASME III and IX	Summer 1967
Replacement Steam Generator (primary and secondary sides)	ASME III Class 1	1989 No addendum

Note:

1. Welded joints tested in accordance with requirements of Article 7, Summer 1966 Addenda.
2. This table reflects original design/construction code information. Refer to UFSAR Section 5.2.2 for additional information on Reactor Coolant System Codes and Classifications.
3. Input Document for Replacement RVCHA Licensing and Safety Evaluation, Babcock & Wilcox Canada, BWC Report No. 068S-LR-01 Rev 2; OM 201.R-0141.001.
4. History Docket for Closure Heads, Customer Spec. # OSS-0279.00-00-003, Babcock & Wilcox Canada, BWC-Cont. 068S, 068S-01.
5. Reactor Coolant piping was requalified to the 1983 ASME code during the Steam Generator Replacement project.

The reactor internals for ONS are discussed in UFSAR Section 4.5.1. They were fabricated prior to Subsection NG of the ASME Code becoming a requirement. The RV internals were designed based on the thermal conditions in the Reactor Coolant System (RCS) Functional Specification. The MUR power uprate conditions are bounded by the RCS Functional Specification. As such, the RV internals remain acceptable under MUR power uprate conditions.

The code design criteria for interfacing systems are identified in UFSAR Section 3.2.2 and Table 3-2. No stress/fatigue analyses were revised, therefore no code of record changed.

IV.1.E **The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.**

RESPONSE:

IV.1.E.i Inservice Inspection Program

10 CFR 50.55a(g), In-service Inspection Requirements, requires the development and implementation of an Inservice Inspection (ISI) Program. The ISI Program is discussed in UFSAR Section 5.2.3.12.4. ASME Class 1, 2 and 3 components are examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI in effect as specified in10CFR 50.55a(g) to the extent practical. The MUR power uprate conditions were reviewed for impacts on the ISI Program. The ISI Program will continue to assess the operational qualification of ASME Class 1, 2, and 3 systems. The Program does not require revision as a result of the MUR power uprate.

IV.1.E.ii Inservice Testing Program

10 CFR 50.55a(f), In-service Testing Requirements, requires the development and implementation of an Inservice Testing (IST) Program. The IST Program establishes performance requirements for pump and valve testing. ONS has developed and implemented an IST Program for pumps and valves per these requirements. The proposed MUR power uprate does not have any impact to the programmatic aspects of the IST Program. It does not change any of the regulatory requirements of the program or in any way change the scope of the program. It does not add or delete any systems or components, since the new LEFM will not be part of the IST Program.

IV.1.E.iii Flow Accelerated Corrosion Program

As a result of plant and industry experience with pipe degradation in process systems, a Flow Accelerated Corrosion (FAC) program was developed at ONS. The purpose of the program is to monitor piping systems that are subject to FAC degradation, and to mitigate pipe wall loss. The FAC program is based on the most current Electric Power Research Institute (EPRI) recommendations and best industry practices.

ONS uses the EPRI CHECWORKS ™ Steam/Feedwater Application (SFA) monitoring software to model operating conditions, material data, and ultrasonic testing (UT) inspection data to provide a calculated estimate of component wear. The thermodynamic changes associated with the MUR power uprate will impact corrosion rates for components located in FAC susceptible systems. All changes required to reflect the MUR power uprate conditions have been incorporated in the CHECWORKS ™ models and the final results and databases have been validated.

Sample results of a wear rate analysis performed for Unit 3 to assess the impact of the MUR power uprate on susceptible FAC components are provided in Table IV.1.E-1 and Figure IV.1.E-1, providing a comparison of the pre-MUR and post-MUR wear rates. Per this analysis, the increase in wear rates due to the MUR power uprate is considered minor and the existing FAC Program is adequate to incorporate the updated predictions. The Unit 3 sample results are considered representative of Units 1 and 2.

Table IV.1.E-1: Sample Results MUR Impact on FAC Wear Rates for Steam Cycle Locations (Unit 3)

Location			Wear Rate Change[1]					Temperature[3]		Steam Quality[4]		Velocity[5]		
No.	Steam Cycle Location	No. of Comps Analyzed	Avg Change (%)	Max Change (%)	Min Change (%)	Avg ECF	Avg Post Wear Rate[2] (mils/yr)	Post Temp (deg F)	Temp Change (deg F)	Post Quality	Quality Change	Post Velocity (f/s)	Velocity Change (%)	Notes
1	Condensate: Condensate Booster Pumps to "F" Heaters	116	1.0%	1.5%	0.9%	1.010	2.2	115.2	0.5	0.000	0.000	12.2	1.9%	
2	Condensate: Condensate Booster Pumps to Feedwater Pump Seals	165	-0.1%	-0.1%	-0.1%	0.999	2.0	115.2	0.5	0.000	0.000	2.1	0.0%	
3	Condensate: "F" Heaters to "E" Heaters	95	1.7%	2.2%	1.7%	1.017	0.9	142.4	0.7	0.000	0.000	10.6	1.9%	
4	Condensate: "E" Heaters to "E" Drain Pump Tie-In	35	1.7%	2.2%	1.6%	1.017	1.9	213.2	0.9	0.000	0.000	11.5	1.9%	
5	Condensate: "E" Drain Pump Tie-in to "D" Heaters	37	1.7%	2.5%	1.6%	1.017	1.4	213.2	0.9	0.000	0.000	10.2	2.0%	
6	Condensate: "D" Heaters to "D" Drain Pump Tie-In	38	1.4%	1.9%	1.3%	1.014	1.2	287.9	1.2	0.000	0.000	10.7	2.0%	
7	Condensate: "D" Drain Pump Tie-In to "C" Drain Coolers	32	1.6%	2.4%	1.5%	1.016	1.5	290.2	1.2	0.000	0.000	9.4	2.1%	
8	Condensate: "C" Drain Coolers to "C" Heaters	28	0.1%	0.1%	0,1%	1.001	1.4	313.0	1.4	0.000	0.000	9.8	2.1%	
9	Condensate: "C" Heaters to Feedwater Pump Suctions	117	1.0%	1.6%	1.0%	1.010	1.3	367.0	1.5	0.000	0.000	11.1	2.2%	
10	Feedwater: Feedwater Pump Discharges to "B" Heaters	91	0.9%	1.5%	0.9%	1.009	1.7	368.4	1.6	0.000	0.000	13.7	2.2%	
11	Feedwater: Feedwater Pump Recircs to Condenser	78	0.9%	1.5%	-1.1%	1.009	14.3	367.2	1.5	0.003	0.000	97.6	2.1%	Location of high predicted wear rate, especially due to flashing occurring downstream of the control valve to the condenser and high average velocity.
12	Feedwater: "B" Heaters to "A" Heaters	46	-15.4%	-15.4%	-15.4%	0.846	2.2	410.3	1.8	0.000	0.000	11.3	2.2%	The small temperature change at this location leads to a significant wear rate change due to location in the FAC Wear Rate Temperature curve.
13	Feedwater: "A" Heaters to SG Inlets	226	2.6%	3.1%	2.5%	1.026	1.5	457.8	2.2	0.000	0.000	14.9	2.3%	In contrast to Location No. 12, a similar change in temperature and velocity leads to an increase in wear rate. In this case the velocity impact outweighs the temperature impact.
14	Bleed Steam: HPT Blead Steam to "A" Heaters	71	N/A	N/A	N/A	N/A	0.0	473.3	3.4	1.000	0.000	N/A	N/A	FAC wear rates for these components were zero both pre-MUR and post-MUR due to superheated operating conditions at both Power Levels.
15	Extraction Steam Reheat: HPT Extraction Steam to 1st Stage Reheaters	70	N/A	N/A	N/A	N/A	0.0	473.3	3.4	1.000	0.000	N/A	N/A	FAC wear rates for these components were zero both pre-MUR and post-MUR due to superheated operating conditions at both Power Levels.
16	Bleed Steam: HPT Bleed Steam to "B" Heaters	8	-1.8%	-1.7%	-1.9%	0.982	3.2	413.7	1.8	0.966	0.001	47.8	1.5%	
17	Bleed Steam: HPT Bleed Steam to "C" Heaters	6	0.3%	0.4%	0.2%	1.003	1.6	373.2	1.6	0.936	0.000	22.9	2.0%	
18	Bleed Steam: LPT Bleed Steam to "D" Heaters	170	N/A	N/A	N/A	N/A	0.0	328.1	0.1	1.000	0.000	N/A	N/A	FAC wear rates for these components were zero both pre-MUR and post-MUR due to superheated operating conditions at both Power Levels.
19	Bleed Steam: LPT Bleed Steam to "E" Heaters	152	5.2%	6.4%	2.0%	1.052	0.6	217.0	0.6	0.933	0.000	0.8	8.1%	
20	Bleed Steam: LPT Bleed Steam to "F" Heaters	60	5.9%	6.6%	4.7%	1.059	1.4	154.0	1.0	0.440	0.001	6.6	3.4%	Location of the greatest average increase in wear rate across the model.
21	Reheater Drains: 1st Stage Reheater Drains to Reheater Drain Tanks	116	3.9%	3.9%	3.9%	1.039	0.6	465.5	2.1	0.000	0.000	2.3	3.2%	

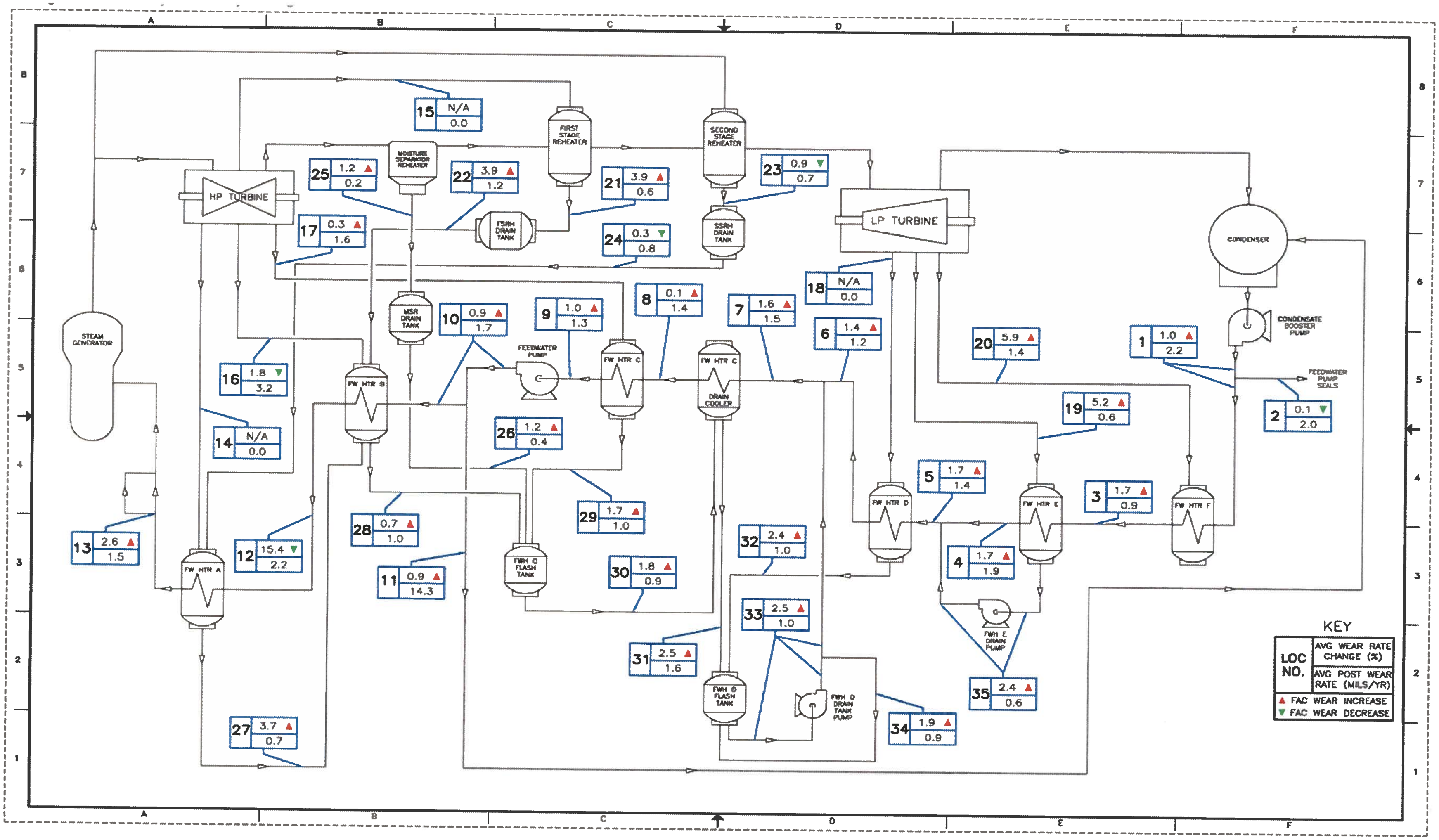
Table IV.1.E-2: Sample Results MUR Impact on FAC Wear Rates for Steam Cycle Locations (Unit 3) (continued)

Location			Wear Rate Change[1]				Avg Post Wear Rate[2] (mils/yr)	Temperature[3]		Steam Quality[4]		Velocity[5]		Notes
No.	Steam Cycle Location	No. of Comps Analyzed	Avg Change (%)	Max Change (%)	Min Change (%)	Avg ECF		Post Temp (deg F)	Temp Change (deg F)	Post Quality	Quality Change	Post Velocity (f/s)	Velocity Change (%)	
22	Reheater Drains: 1st Stage Reheater Drains to "B" Heaters	149	3.9%	4.9%	-8.5%	1.039	1.2	455.3	2.0	0.014	0.000	6.2	3.2%	
23	Reheater Drains: 2nd Stage Reheater Drains to Reheater Drain Tanks	128	-0.9%	-0.9%	-0.9%	0.991	0.7	528.9	0.0	0.000	0.000	1.5	-1.0%	
24	Reheater Drains: 2nd Stage Reheater Drain Tanks to "A" Heaters	127	-0.3%	0.1%	-0.9%	0.997	0.8	493.3	1.1	0.056	-0.002	4.8	-2.5%	
25	MSR Drains: MSR Drains to MSR Drain Tanks	96	1.2%	1.2%	1.2%	1.012	0.2	371.2	1.5	0.000	0.000	1.9	1.1%	
26	MSR Drains: MSR Drain Tanks to "C" Flash Tanks	168	1.2%	1.2%	0.8%	1.012	0.4	371.2	1.5	0.000	0.000	6.0	1.1%	
27	Heater Drains: "A" Heater Drains to "B" Heaters	80	3.7%	4.2%	3.1%	1.037	0.7	419.6	1.8	0.002	0.000	7.6	2.1%	
28	Heater Drains: "B" Heater Drains to "C" Flash Tanks	120	0.7%	1.7%	0.3%	1.007	1.0	382.2	1.5	0.001	0.000	9.1	2.5%	
29	Heater Drains: "C" Heater Drains to "C" Flash Tanks	24	1.7%	1.7%	1.7%	1.017	1.0	373.2	1.6	0.000	0.000	1.2	2.7%	
30	Heater Drains: "C" Flash Tank Drains to "C" Drain Coolers	22	1.8%	1.8%	1.8%	1.018	0.9	372.7	1.5	0.000	0.000	2.8	2.4%	
31	Heater Drains: "C" Drain Cooler Drains to "D" Flash Tanks	84	2.5%	2.6%	1.9%	1.025	1.6	296.4	1.3	0.001	0.000	7.2	2.4%	
32	Heater Drains: "D" Heater Drains to "D" Flash Tanks	25	2.4%	2.4%	2.4%	1.024	1.0	294.0	1.2	0.000	0.000	1.8	2.6%	
33	Heater Drains: "D" Flash Tanks to Condensate Tie-In	101	2.5%	2.6%	1.9%	1.025	1.0	293.7	1.2	0.000	0.000	6.5	2.4%	
34	Heater Drains: "D" Heater Drain Tank Pump Recircs to "D" Flash Tanks	44	1.9%	2.6%	1.3%	1.019	0.9	295.1	1.2	0.000	0.000	10.9	2.4%	
35	Heater Drains: "E" Heater Drains to Condensate Tie-In	149	2.4%	2.5%	1.8%	1.024	0.6	213.2	0.9	0.000	0.000	5.0	2.4%	
	Total Components Analyzed	**3074**												

Notes:

1) Values in GREEN show where FAC has decreased while values in RED show where FAC has increased. In the wear rate change columns negative values are GREEN and positive values are RED.
2) Average Post wear rate based on Pass 2 predictions (calibrated to inspection data).
3) In the temperature change field, values move toward the FAC peak (at ~275 deg F for 1-phase and 300 deg F for 2-phase) are RED while those that move away from the peak are GREEN. Values that are near the FAC peak where the impact is unknown are black.
4) In the quality change field, values that move toward the FAC peak (at ~50%) are RED while those that move away are GREEN. Values that are near the FAC peak where the impact is unknown are black.
5) In the velocity change field, values in GREEN show where velocity has decreased while values in RED show where velocity has increased. FAC wear rates increase with increasing velocities and decrease with decreasing velocities.

Figure IV.1.E-1: Steam Cycle Summary Drawing (Unit 3)



IV.1.F **The discussion should address whether the effect of the power uprate on steam generator tube high cycle fatigue is consistent with NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.**

RESPONSE:

The flow induced vibration (FIV) analyses meet the requirements of the existing design basis and are therefore bounding for MUR conditions.

The FIV and Structural analyses prepared for use in qualifying plugged, sleeved, and stabilized ROTSG tubes meet the requirements of the existing design basis and are therefore bounding for MUR conditions.

Davis-Besse, Unit 1 operates with similar ROTSGs, and has also experienced moderate amounts of tube wear. Davis-Besse operates at a licensed power level of 2817 MWt which bounds the ONS MUR power uprate level of 2610 MWt. The Davis-Besse ROTSGs have been in operation since 2014. They have not experienced excessive tube wear rates or tube-to-tube wear that would indicate that Fluid-Elastic-Instability is active.

The increase in bundle entrance velocity due to the MUR conditions is approximately 2%, which will have an insignificant impact on the wear rate due to foreign objects as compared to the wear rate that would be experienced at pre-MUR conditions. In addition, the small increase in mass flow rate in the steam generators due to MUR conditions is not expected to result in the generation of additional foreign objects or loose parts. Furthermore, the feedwater nozzle design of the ONS replacement steam generators precludes the introduction of large foreign objects into the steam generator. Only foreign objects less than 3/16 inch in diameter can pass through the flow openings in the feedwater nozzles. Based on OTSG operating experience, the rate of tube wear produced by such small foreign objects is low and has not produced wear scars of a size that would challenge the structural integrity of the tubing.

ROTSGs do not operate with blowdown flow during full-power operation. Therefore, operation at MUR conditions will have no impact on blowdown.

NRC Bulletin 88-02 describes an event in which a fatigue failure occurred in a steam generator tube. It is noted that this event occurred in a U-tube steam generator, and that necessary preconditions included denting of the tube at the upper support plate, a high fluid-elastic instability ratio and the absence of effective anti-vibration bar support. This mode of failure is considered implausible in ONS ROTSGs on the basis that:

- The FIV analysis demonstrated an acceptable fluid-elastic instability ratio for ONS ROTSGs,
- OTSG tube support is provided by the broach plates which cannot be mislocated as is possible for U-bend anti-vibration bar supports,
- The ONS broach plates are stainless steel and cannot support "oxide-jacking" leading to tube denting.

Therefore, it is concluded that the existing analyses fully address MUR conditions and the ROTSGs continue to satisfy all original design criteria.

References for Section IV:

IV.1. Letter from T. Preston Gillespie, Jr. (Duke) to U.S. Nuclear Regulatory Commission, dated February 27, 2013, “Oconee Nuclear Station Unit 1, Docket Number 50-269, Unit 1 End of Cycle (EOC) 27 Refueling Outage Inservice Inspection (ISI)” (ML13064A410)

IV.2. Letter from Scott L. Batson (Duke) to U.S. Nuclear Regulatory Commission, dated March 3, 2014, “Oconee Nuclear Station Unit 2, Docket Number 50-270, Unit 2 End of Cycle 26 Refueling Outage Inservice Inspection Summary Report” (ML14066A069)

IV.3. Letter from Scott L. Batson (Duke) to U.S. Nuclear Regulatory Commission, dated August 12, 2014, “Oconee Nuclear Station Unit 3, Docket Number 50-287, Unit 3, End of Cycle 27(3EOC27) Refueling Outage Inservice Inspection (ISI) and Steam Generator Inservice Inspection (SG-ISI) Report” (ML14237A050)

IV.4. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A). EPRI Final Report 1022863, December 11, 2011

IV.5. Nuclear Energy Institute Document, NEI 03-08, Rev. 2, “Guidelines for the Management of Materials Issues,” January 2010

IV.6. Letter from James R. Hall (US NRC) to Scott Batson (Duke Energy), dated June 19, 2015, Issuance of Amendments Regarding Inspection Plan for Reactor Vessel Internals (ML15050A671)

IV.7. Letter from M. S. Tuckman (Duke) to U. S. Nuclear Regulatory Commission, dated July 6, 1998, Subject: “Application for Renewed Operating Licenses Oconee Nuclear Station, Units 1, 2, and 3,” Volume III (ML15254A151)

IV.8. BAW-2241P-A, Revision 2, Fluence and Uncertainty Methodologies

IV.9. Letter from Richard V. Guzman (US NRC) to Scott Batson (Duke Energy) dated February 27, 2014, “Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Revised Pressure-Temperature Limits” (ML14041A093)

IV.10. Framatome Inc. Document 77-3127-002 (ANP-3127, Revision 2), “Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY” (ML13305A121)

IV.11. BAW-10046A, Revision 2, “Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G,” June 1986

IV.12. Framatome Document 43-2192-00 Supplement 1P-A-00-000 (BAW-2192 Revision 0 Supplement 1P-A Revision 0), “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads,” December 2018

IV.13. Framatome Document 43-2178-00 Supplement 1P-A-00-000 (BAW-2178 Revision 0 Supplement 1P-A Revision 0), “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads,” December 2018

IV.14. Framatome Inc. Document 43-1543-04 (BAW-1543, Revision 4), *“Master Integrated Reactor Vessel Surveillance Program,”* February 1993

IV.15. Framatome Inc. Document 43-1543-04_Supplement_7-A-000 (BAW-1543, Revision 4, Supplement 7-A), *“Supplement to the Master* Integrated *Reactor Vessel Surveillance Program,”* March 2018

IV.16. ASTM Standard E185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” American Society for Testing and Materials, Philadelphia, Pennsylvania

IV.17. NRC Regulatory Issue Summary 2014-11, Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components, October 14, 2014 (ML14149A165)

IV.18. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1989 Edition

V Electrical Equipment Design

V.1 **A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:**

RESPONSE:

ONS electrical systems were reviewed. Below is a brief summary of each electrical system. Specific RIS questions are then addressed separately.

The Main Power System

The Main Power System for each unit includes the generator, voltage regulator, isolated phase buses, main step-up transformer and unit auxiliary transformer. The Main Power System generates power, transmits it to the transmission system, and supplies auxiliary power for normal plant operation. The Main Power System continues to have adequate capacity and capability for plant operation with an MUR power uprate, and is bounded by the existing analysis and calculations of record for the plant.

Main Generator

The rating for each unit's main generator is:

- Units 1, 2, 3 - 934 MWe, 1037.937 MVA, 452 MVAR, 0.9 pf

Pre-MUR operating generator output for each unit is:

- Unit 1 - 906.5 MWe, 1023 MVA, 475 MVAR, 0.88 pf
- Unit 2 - 907.3 MWe, 1024 MVA, 475 MVAR, 0.88 pf
- Unit 3 - 914.8 MWe, 1024 MVA, 460 MVAR, 0.89 pf

The expected output for the MUR power uprate for each unit is as follows:

- Unit 1 - 924.6 MWe, 1033 MVA, 460 MVAR, 0.89 pf
- Unit 2 - 924.8 MWe, 1033 MVA, 460 MVAR, 0.89 pf
- Unit 3 - 932.5 MWe, 1037 MVA, 455 MVAR, 0.90 pf

The Main Generator rating is adequate for the pre-MUR unit outputs and will continue to be adequate for the MUR power uprated output. The increases in MWe will result in modest reduction in reactive power. The Main Generator rating is adequate for the current unit outputs and will continue to be adequate for the MUR power uprated output.

Note that the Isolated Phase Bus (IPB) has adequate electrical capacity for the upgrade, but has experienced cooling problems. Those cooling problems are discussed in Section VI.1.C.

Main Step-Up Transformer (MSU)

Units 1 and 2 MSUs are rated at 1000/1120 MVA at 55°C/65°C, 18.1 / 230kV, 3-phase. The Unit 3 MSU is made up of 3 single-phase transformers; each rated 373.333 MVA at 65°C rise, 18.05 / 525kV.

Each MSU receives power from its associated Main Generator and transmits the power to the switchyard. With the Unit 1 and 2 Main Generators operating at MUR power uprate conditions, the associated MSUs will each be loaded to 986.515 MVA. Similarly, with the Unit 3 Main Generator operating at MUR power uprate conditions, its MSU will be loaded to 990.335 MVA. In each case, the load is less than the rating of the MSU.

Auxiliary Transformer (1T, 2T, 3T)

All three UATs are 3-phase, 18.1 / 6.9 / 4.16kV transformers.

Units 1 and 2 (1T, 2T) are each rated for 45/60 MVA.

Unit 3 (3T) is rated for 35 / 70 MVA.

Each UAT has two low voltage windings to serve both 4.16kV and 6.9kV buses. The UATs are sized to supply the full load auxiliaries of one unit as well as the Engineered Safeguard equipment of another unit. Analysis with the current Electrical Transient and Analysis Program (ETAP) model is bounding for loading changes required for achievement of MUR power uprate condition. Therefore, any increase in current flow through the transformers due to MUR uprated conditions remain within transformer ratings.

CT-1, 2, 3 Start-Up Transformers

Start-up transformers CT-1, CT-2, & CT-3 are three winding 230kV / 6.9 kV / 4.16 kV transformers rated for 33.6 MVA. Current operating conditions analyzed using ETAP are already conservative enough to bound any loading increases that will be seen with MUR power uprate conditions; therefore, no additional analysis is required to determine that Start-up Transformer ratings are not impacted by the MUR power uprate. Secondary voltages remain within acceptable limits of less than 105% loaded or less than 110% unloaded.

CT-4 Transformer

The CT-4 transformer is a two-winding 13.2/4.16 kV transformer rated for 12/16/20 MVA. The loading for the ONS units does not increase at the bus level from the existing analysis for either MUR power uprate operation or a LOCA following operation at MUR power uprate conditions. CT-4 transformer ratings remain sufficient for operation at MUR power uprate conditions.

AC Distribution

The following AC distribution systems were reviewed:

4 kVAC Essential Auxiliary Power System,
6.9 kVAC Auxiliary Power System,
600/208 VAC Safety-Related Power System,
600/208 VAC Non-Safety-Related Power System,
120V AC I&C Power System.

The 4kV Essential Auxiliary Power system powers the loads that will have an increase in required power to achieve the uprate. These changes are duplicated for all three units. The 6.9kV, 600/208V Safety-Related and Non-Safety-Related, and 120V systems will experience no loading changes and will

perform their design functions at MUR power uprate conditions. The following motor loading changes are those required to achieve the MUR power uprate:

- **E Heater Drain Pump**

  Worst-case loading with MUR power uprate conditions is 220 bhp per pump. All trains of the Heater Drain System (and all pumps) are operated for normal operating condition. Current ETAP loading for normal conditions is 285 bhp (95% of nameplate). The current ETAP calculation is bounding for this pump. No change is made to the model for representation of MUR power uprate condition.

- **D Heater Drain Pump**

  Worst-case loading with MUR power uprate conditions is 1477 bhp (73.9% of motor nameplate) per pump. Current ETAP model loading for normal conditions is 1700 bhp (85% of nameplate). The current ETAP calculation is bounding for this pump. No change is made to the model for representation of MUR power uprate condition.

- **Hotwell Pumps**

  Worst-case loading for each Hotwell Pump (running all three pumps, representative of actual operating conditions) with MUR power uprate conditions is 650 bhp (65% of motor nameplate). Current ETAP model loading for normal conditions is 650 bhp (65% of nameplate). The current ETAP calculation is bounding for this pump. No change is made to the model for representation of MUR power uprate condition.

- **Condensate Booster Pumps**

  Worst-case loading for each Condensate Booster Pump with MUR power uprate conditions is 1850 bhp (92.5% of nameplate). The Condensate Booster Pump Motors are already modeled conservatively at 100% in the ETAP calculation; therefore, no change is necessary to reflect changes required to achieve MUR power uprate condition.

The current ETAP calculation is bounding for all pumps. No change is made to the model for representation of MUR power uprate conditions. Voltages and currents are still bounded by current calculations. No new load flow analysis, motor starting analysis or short circuit analysis was performed to determine acceptability of loading increases to the affected buses because the plant's current ETAP model already includes sufficient conservatism to bound all loading increases required to achieve the MUR power uprate. Any 4.16kV buses experiencing load increases due to the uprate will not experience unacceptable steady-state voltages, overload or short circuit currents as they remain within the previously analyzed and acceptable loading conditions.

Loads that will increase as a result of the MUR power uprate are limited to certain motors. The motors that will be required to produce additional mechanical power are those that drive the following pumps: Hotwell Pumps, Condensate Booster Pumps, D Heater Drain Pumps, and E Heater Drain Pumps.

The required power to the hotwell pumps for MUR power uprate conditions was determined in the Performance Evaluation of Power System Efficiencies (PEPSE) heat balance evaluation. The hotwell pumps were modeled within the heat balance using the appropriate pump information, such as head and efficiency curves. The heat balance models two hotwell pumps in operation. Note that normal operation is with three pumps in service, which requires the greatest total motor power. However, the power required per motor is greatest with two pumps in service. Using bounding heat balance

conditions, the power required per pump for MUR power uprate conditions is presented in the table below.

As with the hotwell pumps, the required power to the condensate booster pumps for MUR power uprate conditions was determined in the PEPSE heat balance evaluation. The condensate booster pumps were modeled within the heat balance using the appropriate pump information, such as head and efficiency curves. The heat balance models two condensate booster pumps in operation. Using bounding heat balance conditions, the power required per pump for both pre-MUR and MUR power uprate conditions is presented in the table below.

The D heater drain pumps have been de-staged, removing the tenth stage impeller and replacing it with a spacer. Due to this, the existing pump curves do not accurately reflect the pump's head and power performance. Therefore, the required power for these pumps is determined using bounding flows from the heat balance, pressure differentials from the heat balance, and pump efficiency from the existing pump curves. Note that a pump efficiency curve for the de-staged pump does not exist. Pump efficiency is assumed not to significantly change due to the de-staging. Any small change in efficiency is significantly bounded by the conservatism of the bounding flow conditions. The power required per pump for MUR power uprate conditions is presented in the table below.

The power requirements for the E Heater Drain Pumps were determined using the BHP curve for the E Heater Drain Pumps and the E Heater Drain Pump flow rates calculated for pre- and post-MUR power uprate conditions. Two E Heater Drain Pumps were assumed in operation, which corresponds to the normal operating configuration. The pump power required per pump for pre- and post-MUR power uprate conditions is presented in the table below.

The table below provides the changes in BHP of the above pumps due to the MUR power uprate as well as a comparison to the rated motor horsepower.

Table V.1-1

Component	Pre-MUR BHP (per pump)	MUR BHP	Rated Motor HP
Hotwell Pump (2)	700*	700*	1000
Hotwell Pump (3)	650*	650*	1000
Condensate Booster Pump (2)	1850*	1850*	2000
D Heater Drain Pump (2)	1321	1477	2000
E Heater Drain Pump (2)	220*	220*	300

*Note: Because of the small differences in pump flow rates between the pre- and post-MUR power uprate conditions (on the order of 2%) and the minimal slope of the BHP curves in the flow regions of interest, the difference in BHP from pre- to post-MUR power uprate operation is minimal and undiscernible from the BHP curves. Therefore, a difference between the pre- and post-MUR BHP is not identified.

As can be seen in the above table the MUR power uprate does not result in any increase in BHP exceeding rated horsepower of the affected motors.

Protection schemes are determined based upon nameplate data. The existing nameplate ratings of loads supplied by the AC Distribution Systems remain unchanged. Though some motor loads will have increased power requirements (an increase in the percent loading, not an increase in nameplate ratings) due to the MUR power uprate, current will remain below the protective device minimum trip

currents. Protective device settings are therefore bounded by the current analysis based on ETAP model calculations.

Feedwater ultrasonic flow instrumentation is powered from two separate non safety-related power sources from separate power panels. Identical numbers and types of equipment are on each power source and have approximately 5 Amps load.

The Panelboards are fed from 208V MCC's which are fed from 112.5KVA transformers. The additional approximate 600VA load on each transformer is less than 0.5% of the transformer rating. The loads are being supplied from presently spare circuit breakers in each Panelboard. This small load addition to each Panelboard has negligible impact and is acceptable.

All AC distribution systems continue to have adequate capacity and capability for plant operation with an MUR power uprate, and are bounded by the existing analysis and calculations of record for the plant.

DC Distribution

The following DC systems were reviewed:

- 125VDC Keowee Station Power System
- 125VDC SSF Power System
- 125VDC Vital I&C Power System
- 125VDC 230kV Switchyard Power System
- 125VDC 525kV Switchyard Power System
- 250VDC Power System

The MUR power uprate does not affect the capability or operation of the DC distribution systems. LEFM equipment installed in support of the MUR power uprate is connected to DC distribution systems and has been determined to be non-significant and within the capacity margins of the system.

All DC systems continue to have adequate capacity and capability for plant operation after the MUR power uprate, and are bounded by the existing analyses and calculations of record for the plant.

Switchyard Systems

The following switchyard systems were reviewed:

- 230kV Switchyard Power System
- 230kV Switchyard Auxiliary System
- 525kV Switchyard Power System
- 525kV Switchyard Auxiliary System

Units 1 and 2: 230kV Switchyard Systems

The 230kV switchyard connects the Units 1 and 2 main generators and the Keowee Hydro Station generators to the power grid via eight 230kV transmission lines and the autobank transformer. The 230kV switchyard is the connection point for all offsite power sources for the auxiliary power systems for all three units, and it provides the overhead path from the Keowee Hydro Station generators to the auxiliary power transformers. The switchyard is comprised of two 230kV buses, Red and Yellow, twenty-five power circuit breakers (PCBs) and their associated current transformers (CTs), coupling

capacitor voltage transformers (CCVTs), disconnect switches, protective relaying, and auxiliary equipment.

Overhead Lines (Units 1 and 2)

ONS Units 1 and 2 generating units are connected via the Main Step-up Transformers (MSU) to the 230kV transmission lines by the Yellow and Red 230kV buses. The worst-case steady-state load is limited by the maximum current that would be transmitted by the MSU transformers. The maximum rating of both units' MSU is 1120 MVA at 65°C, with a maximum current seen at the 230kV side of the transformer at ~ 2811 Amps per phase.

The MUR power uprate does not impact the MSU rating as discussed above; therefore, the maximum current carried by the overhead lines does not change after the MUR power uprate, and the current ratings are valid.

Motor Operated Disconnect (MOD) Switches (Units 1 and 2)

Motor operated gang switches connect the MSUs and their breakers. Since the MUR power uprate does not impact the transformer rating, the maximum steady state and fault current through the MODs does not change, and the current design remains sufficient.

Unit 3: 525kV Switchyard Systems

The 525kV switchyard transmits power from the ONS Unit 3 Generator to the power grid and provides multiple connection points for power coming into and leaving a central location. The boundaries for the 525kV switchyard include all power equipment including autobank transformer AT-1 and transformer 5T. It is comprised of two electrical buses (Red and Yellow), eight 525kV circuit breakers that are the connection point to the generators, power circuit breakers (PCBs, and motor operated disconnect switches (MODs).

Overhead Lines (Unit 3)

The Unit 3 generating unit is connected via the MSU to the 525kV transmission lines by the Yellow and Red 525kV buses. The worst-case steady-state load is limited by the maximum current that would be transmitted by the MSU transformer. The maximum rating of the three single phase Unit 3 MSU is 373.333 MVA each, with a maximum current seen at the 525kV side of the transformer at ~711 Amps per phase.

The MUR power uprate does not impact the MSU rating as discussed above; therefore, the maximum current carried by the overhead lines does not change after the MUR power uprate, and the current ratings are still valid.

Motor Operated Disconnect (MOD) Switches (Unit 3)

Motor operated disconnect switches are used on either side of all breakers to clear any one of them. This allows the breaker to be cleared without adversely affecting any circuit. A MOD switch is used between the MSU of Unit 3 and its breakers. Since the MUR power uprate does not impact the transformer rating, the maximum steady state and fault current through the MODs does not change, and the current design is still valid.

Power Circuit Breakers (PCBs)

The outputs of Units 1, 2 and 3 generators are connected to the switchyard through PCBs.

Either of the two unit breakers are capable of passing the rated output of the MSUs for each unit. Since the MUR power uprate does not impact the MSU rating, the maximum steady state and fault current through the PCBs does not change, and the current design is still valid.

Autobank Transformer AT-1

Ratings: 22.9 - 240.0Y/138.6 - 525.0Y/303.1kV, 150MVA (tertiary) / 1500MVA at 55°C rise, 168MVA (tertiary) / 1680MVA at 65°C rise

Three single-phase transformers are connected together to form (nominally) a three-phase 1500 MVA, 230kV/525kV autotransformer between the 230kV and 525kV switchyards. AT-1 is connected to PCBs 55 and 56 on the high side, 31 and 33 on the low side. The 22kV (nominal) tertiary winding is used to provide power to the 4.16kV auxiliary power system in the plant through the 5T transformer.

The autobank transformer AT-1 is sized to carry the maximum amount of current that the MSU will provide. The MSU rating is not affected by the uprate; therefore, the increase in MVA is still within the AT-1 rating. There will be no impact due to the MUR power uprate.

Transformer 5T

Ratings: 21.95-4.16Y/2.4kV, 12/16MVA at 65°C rise

Unit Auxiliary Transformer 5T steps down 22kV (nominal) power from switchyard autotransformer AT-1 to the 4.16kV auxiliary power system. Auxiliary system loading will not increase beyond currently analyzed conditions. The current loading will remain acceptable and no impact will be experienced due to the MUR power uprate.

All switchyard systems continue to have adequate capacity and capability for plant operation with an MUR power uprate, and are bounded by the existing analyses and calculations of record for the plant.

V.1.A Emergency Diesel Generators

RESPONSE:

The equivalent emergency diesel generator system for ONS is the Keowee Hydro Station.

The Keowee Emergency Power System is designed to provide a reliable emergency onsite power source for the ONS. The system consists of the Keowee Hydro Station, a 13.8kV underground cable feeder to Transformer CT4, and an overhead 230kV transmission line to the 230kV switching station at ONS which supplies each unit's startup transformer. The Keowee Hydro Station contains two units rated at 87,500kVA each, which generate power at 13.8kV (a common 230kV step-up transformer connects the generators to the transmission line). Each Keowee unit consists of a turbine, generator, exciter, circuit breaker, control equipment, DC control battery, etc.

The Keowee Hydro Units (KHUs) provide emergency electrical power for the plant Engineered Safeguard Features plus selected balance of plant emergency loads. As discussed in UFSAR Section 9.7.3.2, the KHUs also provide alternate QA-1 power to the Protected Service Water System. The MUR power uprate will not change the loading of the Keowee Hydro Units.

The AC Distribution System Evaluation shows that the MUR power uprate will not increase the electrical loading of any component associated with the Keowee Hydro Station. Therefore, Keowee Emergency Power System equipment capacity and capability for plant operations under MUR power uprate conditions are bounded by the loading tables, which are supported by the existing analysis of

record. As a result, the Keowee Emergency Power System will continue to have adequate capacity and capability to operate the plant equipment.

V.1.B Station Blackout Equipment

RESPONSE:

Station Blackout (SBO) is the hypothetical case where all off-site power and both Keowee hydro-electric units are lost. This event is discussed in Section II.1.D.iii.23. As described in UFSAR Section 8.3.2.2.4, the SSF System is credited as the Alternate AC (ACC) power source and the source of decay heat removal required to demonstrate safe shutdown during the required station blackout coping duration of 4 hours for ONS. The SSF System is described in UFSAR Section 9.6. Electrical power is available immediately from the battery systems and within 10 minutes from the SSF diesel generator. The MUR power uprate will have no impact on the design of or the loads supplied from both the battery systems and the SSF diesel generator. Therefore, capacity and capability of electrical power systems for SBO event for plant operation under MUR power uprate conditions are bounded by the load profiles which are supported by the existing analysis of record.

Station blackout systems continue to have adequate capacity and capability for plant operation for the MUR power uprate, and are bounded by the existing analyses and calculations of record for the plant.

V.1.C Environmental Qualification of Electrical Equipment

RESPONSE:

The ONS Environmental Qualification (EQ) Program is guided by the regulations detailed in 10 CFR 50.49, IE Bulletin 79-01 B, Regulatory Guide 1.97, and NUREG- 0737 Supplement 1. Duke Energy has reviewed the ONS EQ program for the MUR power uprate and determined that no EQ Program changes are required as a result of the MUR power uprate.

The review of ONS EQ Program documentation included review of both Duke Energy EQ program-level documents and discrete EQ files/calculations for specific components installed at ONS. This review was conducted to focus on the EQ parameters of temperature, pressure, and radiation, with respect to any potential parameter changes due to the MUR power uprate.

Temperature and Pressure:

Temperature and pressure were evaluated as part of the engineering evaluations for the MUR power uprate. The potential changes in ambient temperatures, system temperatures, system pressures, and potential accident external pressures (high energy line break) and accident temperatures were considered during the review. Radiation was evaluated in a separate review.

The potential impact of the MUR power uprate on ambient plant temperatures was addressed via the HVAC evaluations for the ONS Reactor Building(s), the Auxiliary Building(s), and the Turbine Building(s).

The evaluation for the Reactor Building HVAC System showed that the MUR power uprate would not increase the overall heat load for the Reactor Building, and showed that the Reactor Building ambient temperature would be unaffected by the slight change (approx. 2°F increase) in Main Feedwater system temperature. Therefore, the temperatures used for EQ analysis of Reactor Building components at ONS are unchanged.

The evaluation for the Auxiliary Building HVAC System showed that the 2°F increase in Main Feedwater system temperature due to the MUR power uprate will not affect the heat loading in the

penetration area of the feedwater piping where the HVAC is designed to maintain the overall temperature below 150°F. The evaluation also showed that the MUR power uprate will not impact the HVAC in other areas of the Auxiliary Building.

The Turbine Building at ONS does not contain EQ equipment. Therefore, the MUR power uprate has no impact on qualification analysis relative to the Turbine Building.

The potential impact of the MUR power uprate on system temperature changes was evaluated as part of the MUR power uprate engineering system reviews. The BOP systems review considered the Main Steam System, the Main Feedwater System, and other plant systems (such as Emergency Feedwater, Recirculating Cooling Water, Main Condenser, High and Low Pressure Service Water, etc.). The EQ Evaluation considered potential system temperature changes for plant systems with EQ components (Main Steam and Main Feedwater). The other secondary-side systems evaluated do not contain EQ components, and were not further reviewed for EQ impact. The results of these evaluations showed that Main Feedwater process temperatures changed approximately 2°F (an increase at the inlet and outlet of the MFW pumps, and at FW Heaters A and B), and that Main Steam process temperatures changed approximately 0.5°F (a decrease at the SG outlet and at the Main Steam throttle valve inlet). These slight parameter changes do not affect the qualification of any EQ components at ONS because the temperatures have already been evaluated as not impacting the ambient temperatures of the Reactor and Auxiliary Buildings.

The NSSS systems review considered ONS plant systems related to the Nuclear Steam Supply Systems, such as Reactor Building Spray, Component Cooling, Core Flood, High Pressure Injection, Low Pressure Injection, Reactor Coolant, etc. The review also evaluated instrument & control systems at ONS, such as the Engineered Safety Features Actuation System, the Nuclear Instrumentation & Reactor Protection System, the Integrated Control System, etc. With respect to temperature, the evaluation showed that the RCS hot leg temperature increases approximately 0.4°F due to the MUR power uprate. This slight temperature change was evaluated for environmental qualification impact, and was determined to have no impact on the qualification of EQ components. Overall, the evaluation concluded that there is no impact to system functions due to the MUR power uprate. The evaluation of the I&C NSSS-related systems concluded that impact from the MUR power uprate was negligible, with the exception of minor adjustments needed for Nuclear Instrumentation / Reactor Protection system algorithms, and procedural and administrative updates necessary to account for the increase in core thermal power due to the MUR power uprate and also changes necessary to re-calibrate the Integrated Control System (ICS), which is a non-safety system, is not needed for accident mitigation or accident monitoring, and therefore not subject to EQ requirements. Therefore, the 0.4°F increase in the RCS hot leg temperature has no impact on the EQ Program at ONS.

The potential impact of the MUR power uprate on system pressures was evaluated as part of the BOP and NSSS reviews. The evaluations showed that the Main Steam and Main Feedwater system pressures decrease a slight amount due to the MUR power uprate. These changes were evaluated to have no impact on environmental qualification.

The potential impact of the MUR power uprate on accident external pressures (high energy line break) and temperatures was addressed as part of the BOP evaluation. Overall, there were no constraints identified as part of the HELB program review for the MUR power uprate.

To summarize the evaluation of the temperature and pressure review (due to the MUR power uprate), the BOP systems were determined to show some slight parameter changes, but these minor changes were shown to have no impact on the EQ components at ONS. The evaluation of the NSSS-related systems for temperature and pressure showed that the current design basis analyses (for the pressure and temperature profiles, LOCA and other events) were performed at 102% of core thermal power, which bounds the MUR power uprate. The RCS hot leg temperature increases approximately 0.4°F

due to the MUR power uprate, and RCS pressure is unchanged. There is no EQ impact with respect to temperature or pressure due to the MUR power uprate.

Radiation:

The potential impact of the MUR power uprate on radiation dose was addressed in the EQ evaluation. The evaluation considered EQ component irradiation test information and ONS environmental criteria, and compared the postulated radiation levels (normal and accident values) with available test data. The evaluation also considered the additional 20 years of plant operation (60 years total).

The radiation evaluation was performed with an assumed 2% dose increase (for both normal operation and accident conditions), considered for the remaining licensed lifetime, to account for any increases in radiation dose within the plant due to the MUR power uprate. This assumption was determined to be conservative because the MUR power uprate core thermal power increase 2610 MWt, and because the ONS radiological analyses assume that 100% of the noble gases are released to the recirculating coolant. So, the comparison between postulated dose (total integrated dose, or TID) and EQ test values (irradiation test levels) was performed with conservative plant TIDs.

Some EQ components that were installed later in plant life were evaluated for their expected installed service lives (not the full 60-year period), because they are not subject to the full postulated 60-year TID.

For all the EQ components, either those with a qualified life of 40 years or greater or those that have qualified lives less than 40 years, the radiation evaluation showed the postulated 60-year dose (plus the 2% assumed dose increase to account for the MUR power uprate) was enveloped by the irradiation test values, or the justified dose for the component's actual installed lifetime was enveloped by the irradiation test values, or the component itself was determined to be insensitive to radiation (i.e., comprised of an inorganic material). Therefore, the evaluation of EQ components for any radiation impact due to the MUR power uprate determined that qualification was unaffected. The evaluation determined that all equipment maintained a minimum of 10% margin.

In addition, an evaluation was performed to determine if there were any EQ zones at ONS that are currently classified as mild environments (with TIDs less than 1000 rads) which may be impacted by the 2% assumed dose increase due to the MUR power uprate, and become harsh environments. No EQ zones were identified that will cross this threshold due to the MUR power uprate. The EQ mild zones that were reviewed included the Control Room, the Electrical Equipment Room, and the Turbine Building. The Radwaste Building at ONS was not evaluated because no equipment located in the Radwaste Building is credited with mitigating a LOCA or HELB or used for Post Accident Monitoring. Failure of equipment located in the Radwaste Building will not prevent satisfactory performance of equipment credited with mitigating a LOCA or HELB or used for Post Accident Monitoring. Therefore, equipment located in the Radwaste Building is outside the scope of 10 CFR 50.49.

Summary:

The EQ evaluation for the ONS MUR power uprate review demonstrated that the qualification of EQ components at ONS is not affected by the MUR power uprate, and that temperature, pressure, and radiation parameters remain enveloped by the qualification documentation.

V.1.D Grid Stability

RESPONSE:

A Generation System Impact Study was completed and found to be acceptable for the installation of an additional 45 MWe of generating capacity at Oconee Nuclear Generating Station in Oconee County, SC. This capacity increase is due to the MUR power uprate. The main electrical generators were reviewed for Oconee Units 1, 2, and 3 and were determined to be acceptable for the MUR power uprate.

The power flow cases used in the study were developed from the Duke Energy Carolinas (DEC) internal year projected 2022 summer peak, and winter peak cases. The cases were modified to include 45 MW of additional generating output, which was modeled as an increase in output of 15 MW on each of the three Oconee generating units. To determine the thermal impact on DEC's transmission system, the existing ONS unit generation was increased by 15 MWe per unit. Various faults were placed on the system and their impact versus equipment rating was evaluated. Any significant changes in short circuit current resulting from the additional generation's installation were identified.

A Grid Stability Impact Study for Oconee Units Generation System was performed which addressed four approaches for analysis of the grid with respect to the added generation.

1) Thermal Analysis Study
2) Short Circuit Study
3) Stability Study
4) Reactive Capability Study

Thermal Analysis Study:

The overall thermal analysis study consists of two studies. The first study used the results of Duke Energy (DE) Transmission Planning's annual internal screening as a baseline to determine the impact of additional generation. The annual internal screening identifies violations of the DE Power Transmission System Planning Guidelines and this information is used to develop the transmission asset expansion plan. The annual internal screening provides branch loading for postulated transmission line or transformer contingencies under various generation dispatches. The thermal study results following the inclusion of the additional generation are obtained by the same methods, and are therefore comparable to the annual internal screening. The results are compared to identify significant impacts to the DE transmission system. The evaluation of the addition of 45 MWe individually, along with earlier queued generation, determined there are not any network upgrades required to mitigate thermal loading issues.

The second thermal study utilizes a model that includes the new generation with relevant earlier queued projects and associated known upgrades. The new generation economically displaces DEC Balancing Authority Area units. Transmission capacity is available as long as no transmission element is overloaded under N-1 transmission conditions. The thermal evaluation only considers the base case under N-1 transmission contingencies to determine the availability of transmission capacity. The model uses transmission capacity on an “as available” basis; therefore, adverse generation dispatches that would make the transmission capacity unavailable are not identified. If the full output of the additional 45 MWe generating facility cannot be delivered at the time of the study, the study identifies the maximum allowable output does not require additional network upgrades. The study determined that no network upgrades are required to add 45 MWe of generating capacity.

Short Circuit Study

Short circuit analysis was not performed because the additional generation output of the existing generators will not affect the fault duty. The impedance of the machine has not changed, and thus there is no impact to the fault duty of the system.

Stability Study

Stability studies are performed using a Multiregional Modeling Working Group dynamics model that has been updated with the appropriate generator and equipment parameters for the additional generation. A projected 2022 Summer Peak model was used for this study. The case was modified to turn off some existing generation to offset the additional generation. The power flow portion of the interconnection request did not identify any transmission upgrades associated with the addition of the additional generation that needed to be added to the dynamics case. NERC TPL-001-4 Planning Events and Extreme Events were evaluated.
There are no contingencies that caused angular or voltage stability concerns in the system due to the additional generation requested at the Customer’s generating facility.

With the assumptions and models used in this study, the planned Oconee MUR power uprate will not negatively impact the stability of the DE transmission system. Any changes to assumptions or models may change these results.

Reactive Capability Study

Reactive Capability is evaluated by modeling a facility’s generation and step-up transformers (GSU’s) at various taps and system voltage conditions. The reactive capability of the facility can be affected by many factors including generator capability limits, excitation limits, and bus voltage limits. The evaluation determines whether sufficient reactive support will be available at the Connection Point based on the requirements set forth in DE’s Facilities Connection Requirements (FCR) for generation connected to the Transmission System.

The level of reactive support supplied by the units has been determined to be acceptable with the additional generating output. Evaluation of MVAR flow and voltages in the vicinity of Oconee Nuclear Station indicates adequate reactive support exists in the region. Refer to Figures V.1.D-1 and V.1.D-2 below.

Conclusion

Based on the above studies, Duke Energy has determined that the MUR power uprate will have no significant effect on grid stability or reliability and no modifications to the transmission system are required.

Figure V.1.D-1: Reactive Operating Area (Units 1 and 2)

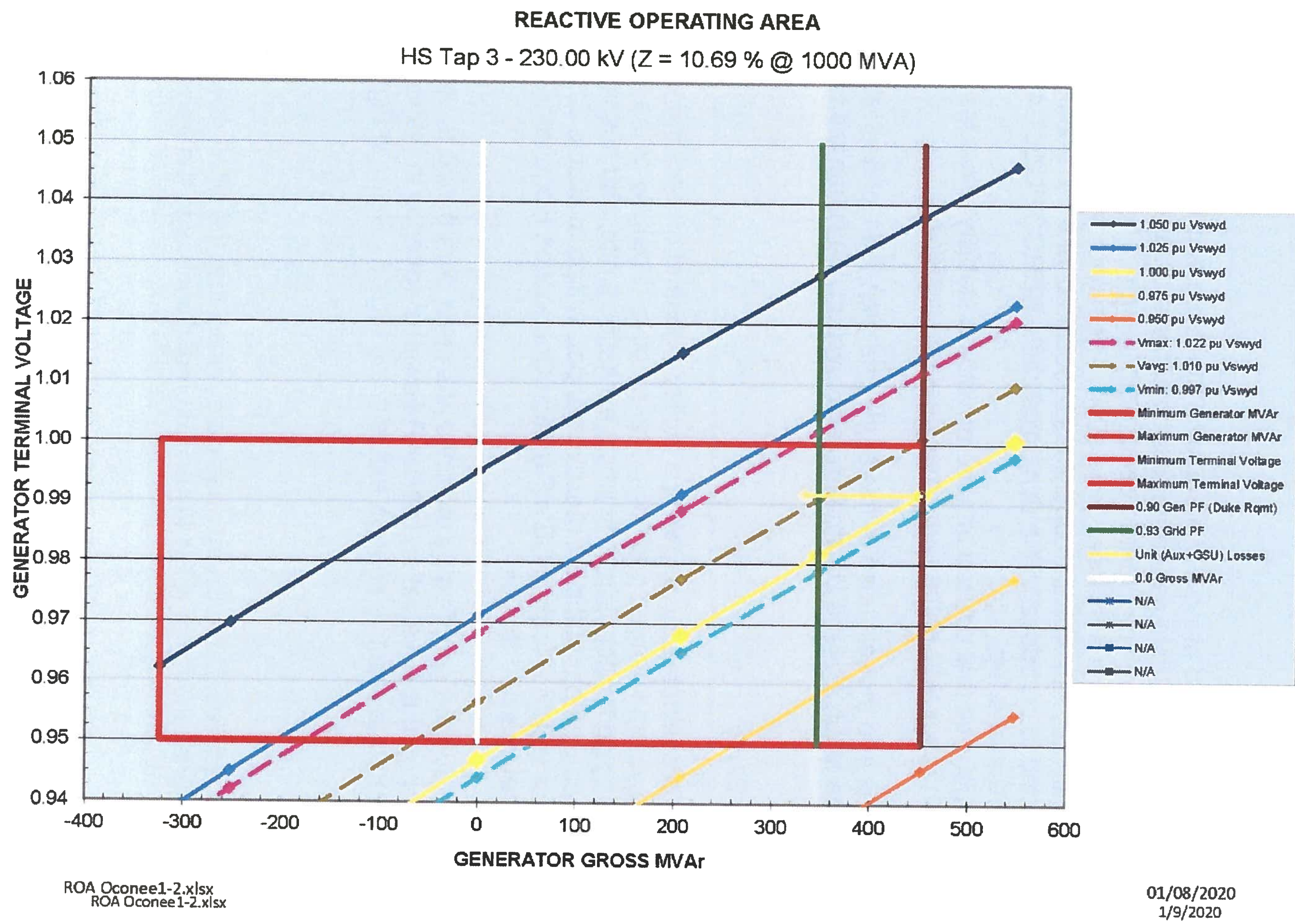
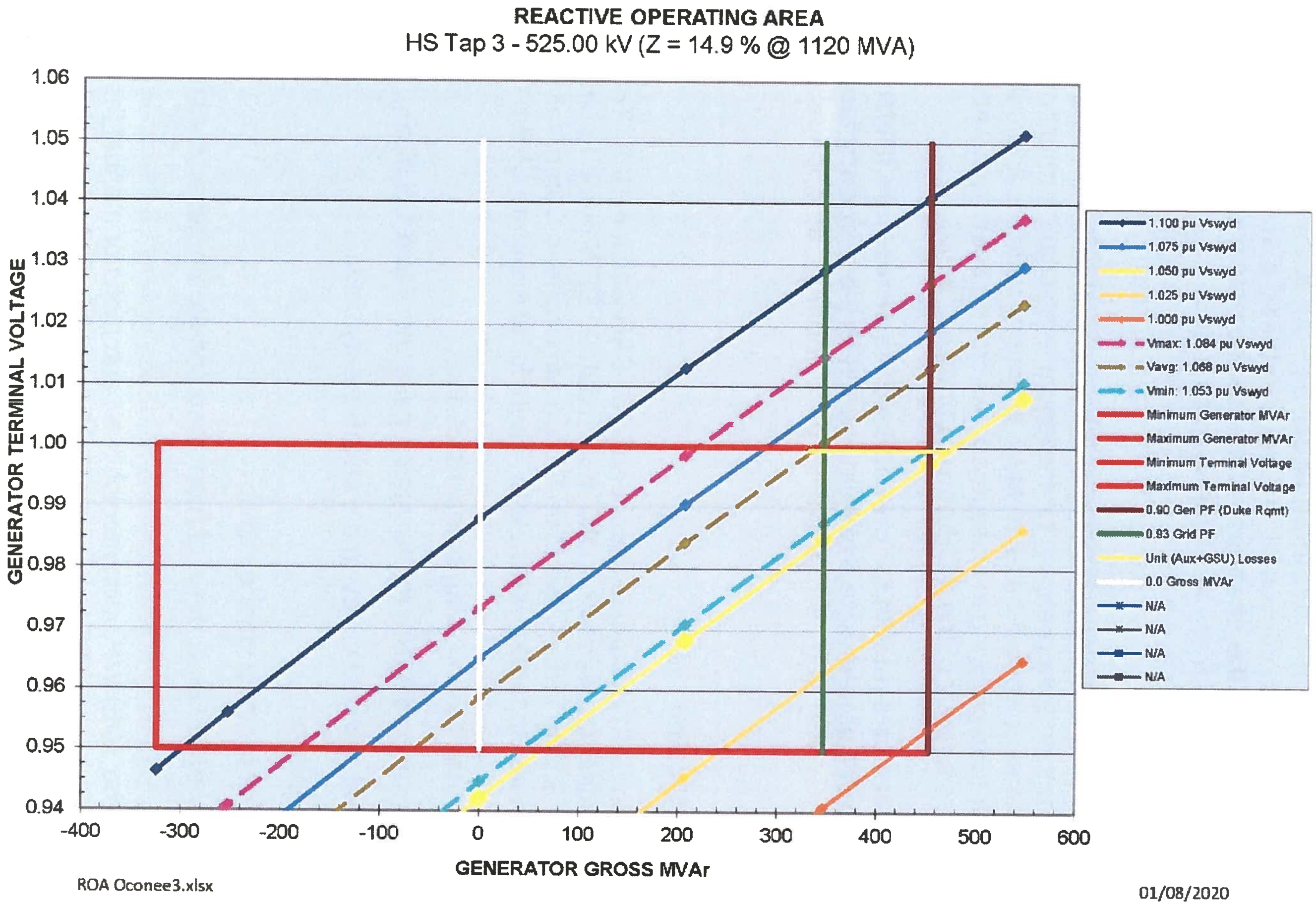


Figure V.1.D-2: Reactive Operating Area (Unit 3)



ROA Oconee3.xlsx

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VI System Design

VI.1 A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems:

RESPONSE:

Table IV-1 contains a summary of the BOP operating parameters before and after the MUR power uprate. MUR power uprate data is shown for maximum analytical thermal power of 2619 MWt (102% of 2568). Licensed thermal power will be 2610 MWt. Summer condenser circulating water conditions and 10% steam generator tube plugging are assumed. As can be seen from Table IV-1, the operating parameter changes as a result of the MUR power uprate are small.

VI.1.A NSSS interface systems for pressurized-water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling-water reactors (BWRs) (e.g., suppression pool cooling), as applicable

RESPONSE:

VI.1.A.i Main Steam

The ONS Main Steam (MS) System is described in UFSAR Section 10.3. The MS System includes not only piping from the steam generators to the main turbines, EFW pump turbines and other loads, but also the Main Steam Safety Valves, the Main Steam Atmospheric Dump Valves, the Turbine Bypass valves, and the Moisture Separator Reheaters. There is no separate Steam Dump system at ONS.

The MS System performs the following safety functions:

- Provide overpressure protection for the steam generators and MS piping,
- Provide decay heat removal via the main steam safety valves,
- Provide MS line isolation,
- Provide decay heat removal via the ADVs,
- Prevent the uncontrolled blow down of more than one steam generator in the event of a MS line rupture,
- Isolate MS from the TDEFWP upon receipt of an automatic or manual feedwater isolation signal,
- Minimize Containment Temperature Increase Due to a Main Steam Line Rupture within Containment,
- Provide steam to the EFW pump turbine,
- Establish containment boundary,
- Provide a fission product barrier sufficient to meet 10 CFR Part 20 public dose limits during normal operation and 10 CFR Part 100 dose limits during design basis event mitigation.

A comparison of the operating conditions for the 2610 MWt MUR power uprate to the current 2568 MWt conditions demonstrates that the Main Steam System has sufficient design and operational margin to

accommodate the MUR power uprate with one exception. By letter dated April 4, 2012, Duke identified that the pressure in the non-safety related Cross Around Piping (CAP) between the high pressure and low pressure turbines, including the moisture separator reheaters (MSRs), exceed the ASME Code transient overpressure rating in the event the Combined Intermediate Valves (CIVs) were inadvertently closed while the turbine stop valves and turbine throttle valves remain open. This condition exists due to sizing of the relief valves and their discharge lines. There is sufficient margin in the piping and equipment to allow a design pressure re-rate. This is identified as a Regulatory Commitment in Attachment 1.

Otherwise, the MS System MUR power uprate conditions remain bounded by the design basis of record.

VI.1.A.ii Main Turbine-Generator

As discussed in UFSAR Section 10.2, the turbine-generator converts the thermal energy of steam produced in the steam generator into mechanical shaft power and then into electrical energy. The turbine-generator consists of a tandem (single shaft) arrangement of a double flow, high pressure turbine and three identical double-flow low pressure turbines driving a direct-coupled generator at 1800 rpm.

The main electrical generators were reviewed at each of the ONS units and it was determined that the electrical generators are acceptable for the MUR power uprate. The increase of MWe due to the MUR power uprate can be accommodated within the present generator nameplate ratings and will result in modest reduction in available reactive power output. A summary of the generator design parameters compared to the MW/MVAR loading before and after the MUR power uprate is given below in Table VI.1-1. The electrical generators are therefore acceptable for the MUR power uprate.

Table VI.1-1: Generator Design Parameters

<table>
<tr><th></th><th colspan="2">ONS Unit 1</th><th colspan="2">ONS Unit 2</th><th colspan="2">ONS Unit 3</th></tr>
<tr><td colspan="7">Turbine nameplate rating: 1037.937 MVA, 0.90 PF</td></tr>
<tr><td rowspan="2">Equivalent nameplate rating</td><td>MW</td><td>MVAR</td><td>MW</td><td>MVAR</td><td>MW</td><td>MVAR</td></tr>
<tr><td>934</td><td>452</td><td>934</td><td>452</td><td>934</td><td>452</td></tr>
<tr><td>Pre-MUR Values</td><td>906.5</td><td>475</td><td>907.3</td><td>475</td><td>914.8</td><td>460</td></tr>
<tr><td>Post-MUR values</td><td>924.6</td><td>460</td><td>924.8</td><td>460</td><td>932.5</td><td>455</td></tr>
</table>

The turbine-generator was reviewed and found to be acceptable for the MUR power uprate level and the unit design rating of 1038 MVA.

VI.1.A.iii Condensate and Main Feedwater

The Condensate and Main Feedwater Systems are described in UFSAR Section 10.4.6. Three motor-driven hotwell pumps deliver condensate from the condenser hotwell through the condensate polishing demineralizers, the condensate and other coolers to the suction of the condensate booster pumps. Three motor-driven condensate booster pumps deliver condensate through four stages of feedwater heating to the main feedwater pumps. Two steam turbine-driven main feedwater pumps deliver feedwater through two high pressure heaters to a single feedwater distribution header where feedwater is divided into two single lines to the steam generators.

A comparison between operating requirements for MUR power uprate conditions and current conditions demonstrates that the Condensate and Main Feedwater System has sufficient design and operational margin to accommodate the MUR power uprate. The MUR power uprate conditions remain bounded by design as described in the Oconee UFSAR.

VI.1.A.iv Emergency Feedwater

The Emergency Feedwater (EFW) System, described in UFSAR Section 10.4.7, provides sufficient feedwater supply to the steam generators of each unit during events that result in a loss of Main Feedwater. EFW removes energy stored in the core and primary coolant. The accident analyses crediting EFW were evaluated at 102% of 2568 MWt and bound the MUR power uprate.

The current design and capabilities of the Emergency Feedwater System remain bounding for the MUR conditions. New MUR power uprate conditions do not impose any necessary changes to system design flow rates, volumes, temperatures or pressures.

VI.1.B Containment Systems

RESPONSE:

The containment systems are provided to limit offsite releases following a Design Basis Accident. These systems include the containment, containment isolation system, Reactor Building Spray and Reactor Building Cooling System. As indicated in Section II above, the existing analyses are shown to remain valid. As such, these systems are not impacted by the MUR power uprate.

VI.1.B.i Reactor Building Spray System

The Reactor Building Spray (BS) System is discussed in UFSAR Section 6.2.2 and consists of two 100% capacity trains that take suction from either the Borated Water Storage Tank (BWST) or from the RB Sump. Each train includes a BS pump with associated valves and piping leading to an array of spray nozzles provided in a header inside the upper area of containment.

The BS System is designed to perform the following functions following a Design Basis Accident (DBA) at ONS:

1) Remove heat from the Reactor Building atmosphere to reduce containment pressure and temperature,

2) Remove the iodine fission product from the RB atmosphere.

There will be no change to the ability of the BS System to perform these safety related functions as a result of implementing the MUR power uprate. Design parameters have been analyzed to envelop operating conditions for the system.

The BS System will continue to perform its safety related functions of containment heat removal and iodine removal after the MUR power uprate. The analyses of record for main steam line breaks and small and large break LOCAs were performed at 2619 MWt (102% of 2568 MWt) and therefore bound the flow, temperature and pressure experienced by the BS System and its components following the MUR power uprate. There is no impact to this system due to the MUR.

VI.1.B.ii Reactor Building Cooling System

The Reactor Building Cooling (RBC) System is described in UFSAR Section 6.2.2 and removes heat from the RB atmosphere following an accident.

The RBC System will continue to perform its safety related functions of heat removal after the MUR power uprate. The RCS design flow, temperature, and pressure prior to and following a DBE are unchanged. The existing Reactor Building Cooling analyses were performed at 2619 MWt (102% of

the original core thermal power of 2568 MWt); therefore, flow, temperature, and pressure experienced by the RBC System and its components will be bounded for a MUR power uprate.

VI.1.B.iii Containment Isolation

The safety function of the penetrations and hatches is to maintain containment integrity under accident conditions. As indicated in Section II of this enclosure, the transients associated with accidents continue to be maintained within design limits. As such, these systems are not impacted by the MUR power uprate.

VI.1.C Safety-Related Cooling Water Systems

RESPONSE:

VI.1.C.i Component Cooling System

The Component Cooling (CC) System is described in UFSAR Section 9.2.1 and provides closed loop cooling water to various heat exchangers located inside the Reactor Building. The heat that is transferred to the CC water from these heat exchangers is in turn transferred from the CC water to the Low Pressure Service Water (LPSW) System in the Component Coolers.

The CC System's only safety function is to provide containment isolation to ensure that RB atmosphere leakage is minimized during a Design Basis Accident (DBA) at ONS.

The CC System will continue to perform its safety function of containment isolation. The margin between the design and operating temperatures and pressures is large enough to allow for the increase in power level from the current power level to the MUR power level. There is no impact to this system due to the MUR.

VI.1.C.ii Condenser Circulating Water

The Condenser Circulating Water (CCW) System is described in UFSAR Section 9.2.2 and consists of four pumps per unit, which take water from Lake Keowee via the intake canal to supply plant systems that use raw water.

The CCW system performs the following safety functions:

- Provide a suction source for the low pressure service water (LPSW) pumps during normal operations and emergencies. Ensure suction is provided to LPSW Pumps during loss of power to all CCW Pumps.
- Provide a suction source for the PSW booster pump via water in the Unit 2 inlet piping. The CCW System is capable of transferring water from all three units' CCW inlet and discharge piping to support 30 days of decay heat removal during a loss of lake event.
- Provide a suction source for SSF ASW System via water in the Unit 2 CCW inlet piping. Ensure flow paths are maintained for makeup to CCW piping via SSF submersible pumps.
- Provide a suction source for HPSW Pumps to support fire suppression for 4 hours during a fire.

The MUR power uprate will have no impact on the LPSW suction source. Suction to the LPSW pumps is provided from connections to the CCW inlet crossover piping. The fluid conditions in this crossover are not impacted by the MUR power uprate conditions.

The MUR power uprate will have no impact on the PSW System suction source. Suction to the PSW booster pump is provided from a connection to the CCW Unit 2 inlet piping. The fluid conditions in the inlet piping are not impacted by the MUR power uprate conditions. The water available to the PSW booster pump in the CCW piping will continue to last approximately 30 days because the volume available in the piping is unchanged due to the MUR power uprate and the volume required is based on decay heat loads assuming 102% of 2568 MWt.

The MUR power uprate will have no impact on the SSF ASW System suction source. Suction to the SSF ASW System is provided from a connection to the CCW Unit 2 inlet piping. The fluid conditions in the inlet piping are not impacted by the MUR power uprate conditions. Also, the CCW System configuration is not impacted by the MUR power uprate; therefore, the CCW System can still ensure a flow path to the SSF submersible pumps is maintained.

The MUR power uprate will have no impact on HPSW suction source. Suction to the HPSW Pumps is provided from connections to the CCW inlet crossover piping. The conditions of the fluid in this crossover are not impacted by the new MUR power uprate conditions. Also, the CCW System configuration is not impacted by the MUR power uprate; therefore, still allowing the CCW System to supply suction to the HPSW pumps for a minimum of two hours.

A comparison between operating conditions for the 2610 MWt MUR power uprate and the current 2568 MWt conditions demonstrates that the Condenser Circulating Water System has sufficient design margin to accommodate the MUR. The safety functions of the Condenser Circulating Water System have been determined not to be adversely impacted by the MUR. The systems remain bounded by the existing analyses of record.

VI.1.C.iii High Pressure Service Water

The High Pressure Service Water System (HPSW) is described in UFSAR Section 9.2.2 and supplies raw lake water for fire protection and cooling/sealing of various loads.

The HPSW system performs the following safety and fire protection functions:

- Prevent air in-leakage from air binding the low pressure service water pumps if the elevated water storage tank is depleted.
- Provide a source of water for fire suppression systems. This is a regulatory requirement.
- Provide fire protection during all plant conditions, for both safety and non-safety related SSCs.

The safety and fire protection functions of the High Pressure Service Water System will not be adversely impacted by the MUR. The system remains bounded by the existing analysis of record.

VI.1.C.iv Low Pressure Service Water

The LPSW System is described in UFSAR Section 9.2.2 and is designed to provide cooling water for normal and emergency services throughout the station. Oconee Units 1 and 2 share three LPSW pumps while Unit 3 has two LPSW pumps.

The LPSW system performs the safety-related function of providing cooling to the low pressure injection coolers, the high pressure injection pump motor bearing coolers, the motor driven EFW pump motor air cooler, the reactor building cooling units, and the siphon seal water headers.

New MUR power uprate conditions do not impose any necessary changes to system design and capabilities of the LPSW System and components remain bounding for the MUR power uprate conditions.

VI.1.C.v Standby Shutdown Facility Auxiliary Service Water (SSF ASW)

The Standby Shutdown Facility is described in UFSAR Section 9.6. The ASW portion of the SSF System is a high head, high volume system designed to provide sufficient steam generator inventory for adequate decay heat removal for all three units. The Unit 2 CCW piping serves as the supply source for the SSF ASW System. The SSF, which includes SSF ASW, serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain Mode 3. The SSF is capable of maintaining all three units at Mode 3 for 72 hours.

The SSF ASW system has no functions related to the design basis events described in Chapter 15 of the UFSAR. For other events described in UFSAR Section 9.6, the mitigation functions of the SSF ASW system are:

- Serve as a backup to the emergency feedwater (EFW) system for events that result in a loss of all EFW and for a single failure that renders condenser hotwell inventory unavailable for the EFW system,
- Mitigate a turbine building flood by providing a source of water from the Unit 2 CCW inlet piping for SG secondary side cooling, drive the SSF-CCW suction line air ejector, and use the submersible pump to replenish the Unit 2 CCW inlet pipe with raw water for the SSF.

The current design and capabilities of the SSF ASW System and components remain bounding for MUR power uprate conditions. New MUR power uprate does not impose any necessary changes to system design flow rates, volumes, temperatures or pressures.

VI.1.C.vi Protected Service Water System

As described in UFSAR Section 9.7, the Protected Service Water (PSW) System is designed as a standby system for use under emergency conditions. The PSW System design includes a dedicated power system and independent control functions. The PSW system provides additional “defense-in-depth” protection by serving as a backup to existing safety systems. The PSW System is provided as an alternate means to achieve and maintain safe shutdown conditions for one, two or three units following certain postulated scenarios. The PSW System reduces fire risk by providing a diverse power supply to power safe shutdown equipment in accordance with the NFPA 805 safe shutdown analyses. The PSW System requires manual activation and can be activated if normal emergency systems are unavailable.

The PSW System is not adversely impacted by the MUR power uprate. PSW can continue to perform all required functions post-MUR while remaining within its current design limits.

VI.1.C.vii Recirculated Cooling Water System:

The Recirculated Cooling Water (RCW) System is described in UFSAR Section 9.2.2 and supplies corrosion-inhibited closed-loop cooling water to various primary and secondary components in the Auxiliary and Turbine Buildings. Major components within the RCW System include surge tanks, RCW pumps, RCW heat exchangers, and Spent Fuel Coolers. The RCW System also includes several minor heat exchangers which provide cooling to both primary and secondary components in the Auxiliary and Turbine Buildings.

The RCW System performs no safety function. The RCW System performs a risk significant function for decay heat removal from the spent fuel pool while the core is offloaded. The spent fuel pool coolers are analyzed at 102% of 2568 MWt.

The RCW System is capable of supporting the MUR power uprate design conditions. The RCW System design flow rate capacity provides adequate design flow to the components within the system at both current and MUR power uprate conditions. The RCW System cools one Isolation Phase Bus (IPB) Air Cooler for each unit. These coolers cool the air that circulates across the Isolation Phase Buses. The IPB ventilation systems for ONS Units 1, 2, and 3 do not meet the original nameplate design flow which correlates to issues for IPB cooling capacity. During periods with elevated outdoor temperature, the cooling system is not capable of providing the cooling necessary to remove the IPB resistance heating. This condition requires that Duke Energy monitor the IPB temperatures and must either provide supplemental cooling or limit the maximum thermal power. Duke Energy plans to upgrade the IPB cooling capability to eliminate this issue.

VI.1.D Spent Fuel Pool Storage and Cooling Systems

RESPONSE:

VI.1.D.i Refueling System (RFS)

The Refueling System (RFS) is described in UFSAR Section 9.1 and consists of plant facilities for storing both new and spent fuel as well as a means for transferring fuel to and from the RB from the Spent Fuel Pools (SFP). The RFS does not perform a safety related function with respect to safe shut down of any of the units at ONS. Because the RFS equipment handles nuclear fuel with the potential to release radioactive fission products if damage occurs from a fuel handling accident, the system is considered “risk significant” from the standpoint of offsite dose limits.

The RFS will continue to perform its risk significant functions of storing new and spent fuel in the SFPs and transporting fuel into and out of the RB. The existing analysis for determining radiation levels of spent fuel was performed at 2619 MWt (102% of 2568 MWt). This analysis bounds radiation levels to be encountered by the fuel storage racks at the MUR power level. Spent fuel being stored in the SFP after being irradiated at the higher power level associated with the MUR power uprate will be maintained in the storage racks in a subcritical condition. There is no impact to this system due to the MUR power uprate.

VI.1.D.ii Spent Fuel Cooling (SF) System

The Spent Fuel Cooling (SF) System is described in UFSAR Section 9.1.3 and is composed of six pumps, six heat exchangers, filters, valves, and interconnecting piping whose function includes cooling, purifying and maintaining water level in the spent fuel pools and the refueling canal.

The SF System does not perform a safety related function; however, it is credited with meeting the Extensive Damage Mitigation Strategy commitment of Section H of the licenses for ONS Units 1, 2, and 3.

The SF system will continue to perform its risk significant functions of spent fuel decay heat removal and SFP inventory control after the MUR power uprate. Analysis demonstrates that the increase in SFP heat load resulting from fuel assumed to have been irradiated at maximum thermal power of 2619 MWt is still within the design parameters of the SF System and its components following the MUR power uprate. There is no impact to this system due to the MUR power uprate.

VI.1.E Radioactive Waste Systems

RESPONSE:

The radioactive waste management systems are described in UFSAR Chapter 11. A review was made of these systems to ensure that these systems would not be affected by the MUR. Based on the system descriptions, functions, and relationships to other systems, the following radioactive waste disposal systems will not be affected by the MUR.

The Gaseous Waste Disposal System (GWD) contains waste gases. GWD failure is addressed in safety analyses. The GWD containment isolation function does not directly or indirectly interface with the steam cycle and therefore is not impacted by the MUR power uprate.

The Liquid Waste Disposal (LWD) System piping provides pressure boundary piping and containment isolation functions for mitigating events. The system also is credited to store and minimize leakage of radioactive fluid to the environment. With no direct interface with the steam cycle, these system functions are unaffected by the MUR power uprate.

The Solid Waste Management System is designed to contain solid radioactive waste materials as they are produced in the station, and to provide for their storage and preparation for eventual shipment to an NRC or Agreement State Licensed offsite disposal facility. The Solid Waste Management System has no direct interface with the power cycle, and therefore, the MUR power uprate will have no impact on this system.

VI.1.F Engineered Safety Features (ESF) Heating, Ventilation, and Air Conditioning Systems

RESPONSE:

VI.1.F.i Control Room Ventilation and Air Conditioning Systems

The Control Room Ventilation and Air Conditioning Systems are described in UFSAR Section 9.4.1. There are two separate control room ventilation systems (CRVS) (one for Units 1 and 2 and a separate CRVS for Unit 3). The WC system provides chilled water to ensure the heat loads, considered vital loads, are removed from the areas served by the CRVSs when air is circulated via the respective cooling coils of the associated air handling units. There is one chilled water system that serves all three units. The control room ventilation systems provide cooling to other areas besides the control rooms. They provide cooling to the cable rooms, the electrical equipment rooms and areas designated as the control room zones.

The control room ventilation systems have a safety function to provide cooling and filtration to the operators, and equipment in the control rooms and associated areas.

A comparison between operating conditions for the 2610 MWt MUR power uprate conditions and the current 2568 MWt operating conditions demonstrates that the control room ventilation systems and the chilled water system have sufficient design and operational margin to accommodate the MUR. The systems remain bounded by the existing analyses of record.

VI.1.F.ii Spent Fuel Pool Area Ventilation System

The Spent Fuel Pool (SFP) Ventilation System is discussed in UFSAR Section 9.4.2 and is designed to maintain a suitable environment for the operation, maintenance, and testing of equipment and for personnel access. Two methods of exhausting air from the Fuel Pool Area are provided, a filtered exhaust system and an unfiltered exhaust system. Normal operation is with the unfiltered system in operation. In the filter mode, the Fuel Pool Area ventilation air passes through a filter train consisting of

prefilters, high efficiency particulate (HEPA) filters, charcoal filter and two 100 percent vane axial fans. The UFSAR Chapter 15 analysis of a fuel handling accident in the spent fuel pool does not credit filtration by the SFP Ventilation System. A comparison of the operating conditions for the 2610 MWt MUR power uprate to the current 2568 MWt conditions demonstrates that the SFP Ventilation Systems have sufficient design and operational margin to accommodate the MUR. The systems remain bounded by the existing analyses of record.

VI.1.F.iii Auxiliary Building Ventilation System

The Auxiliary Building (AB) Ventilation System is discussed in UFSAR Section 9.4.3 and is designed to provide a suitable environment for the operation, maintenance and testing of equipment and for personnel access. The AB ventilation System is not credited during accident conditions.

VI.1.F.iv Reactor Building Purge System

The Reactor Building (RB) Purge System is discussed in UFSAR Section 9.4.5. The RB Purge system provides the RB with fresh air during outages to reduce airborne contaminant levels inside the Reactor Building. The RB Purge system is only used in MODES 5 and 6 or when the fuel has been completely offloaded from the reactor vessel. During MODES 1, 2, 3, and 4, the RB Purge system containment penetrations are required to be sealed closed thereby prohibiting operation of the system.

The only safety function of the RB Purge system is containment isolation. The Reactor Building Purge System will continue to operate to perform its safety related functions of maintaining containment isolation after the MUR power increase. The containment post-LOCA conditions (temperature and pressure) which were performed at 2619 MWt (102% of the original core thermal power of 2568 MWt) are unchanged. There is no adverse impact to this system due to the MUR power uprate.

VI.1.F.v Reactor Building Cooling System

The Reactor Building Cooling System is discussed in Section VI.1.B.ii.

VI.1.F.vi Reactor Building Penetration Room Ventilation System

The Reactor Building Penetration Room Ventilation System is discussed in UFSAR Section 9.4.7.

The system is actuated by the ES System. As a result of the adoption of the Alternate Source Term, the Reactor Building Penetration Room Ventilation filters are no longer credited. A comparison of the operating conditions for the 2610 MWt MUR power uprate to the current 2568 MWt conditions demonstrates that the Reactor Building Penetration Room Ventilation System has sufficient design and operational margin to accommodate the MUR. The system remains bounded by the existing analyses of record.

VII Other

VII.1 A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.

RESPONSE:

The engineering change process ensures that all discipline, system, and program impacts including updates to emergency and abnormal operating procedures, Time Critical Operator Actions (TCOAs), control room controls, HMI displays (including the safety parameter display system) and alarms, the control room plant reference simulator, and the operator training program are captured when a modification occurs. Therefore, all modification packages identified as affected by the MUR power uprate will update these items. The design change process ensures any impacted normal, abnormal and emergency operating procedures having operator actions are revised prior to the implementation of the MUR power uprate if required. An evaluation was performed of the operator actions and no impacts were identified.

TCOAs are associated with the mitigation of postulated events. These actions must be performed in a specified time in order to assure the plant complies with assumptions made during the analysis of these postulated events. The TCOAs were evaluated individually in system evaluations and against the ONS licensing analyses presented in Section II of this enclosure to ensure they remain bounded. All the TCOAs remain unchanged following the MUR power uprate.

VII.2 A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

VII.2.A Emergency and Abnormal Operating Procedures

RESPONSE:

The proposed MUR power uprate will be implemented under the administrative controls of the ONS LAR and design change processes which provide the administrative controls relevant to identifying impacted procedures, controls, displays, alarms, the Operator Aid Computer (which includes the Safety Parameter Display System), and other operator interfaces, the simulator, and training. The design change process ensures any impacted emergency and abnormal operating procedures are revised prior to the implementation of the MUR power uprate.

VII.2.B Control Room Controls, Displays (including the safety parameter display system) and alarms

RESPONSE:

A review of plant systems has indicated that only minor modifications are necessary. ONS follows established engineering and licensing procedures as noted in Section VII.2.A to ensure the necessary minor modifications are installed prior to implementing the proposed MUR. In addition, an “LEFM System Trouble” alarm window will be added to the control room alarm panel to alert the operator when there is a problem with the LEFM. This is included as a Regulatory Commitment in Attachment 1.

VII.2.C Control Room Plant Reference Simulator

RESPONSE:

A review of the plant simulator will be conducted, and necessary changes made, prior to implementing the MUR. The MUR is being implemented under the administrative controls of the ONS LAR and design change processes which provide the administrative controls relevant to identifying impacted procedures, controls, displays, alarms, the Operator Aid Computer (which includes the Safety Parameter Display System), and other operator interfaces, the simulator, and training. The design and licensing change processes ensure any impacted emergency and abnormal operating procedures are revised prior to the implementation of the power uprate. As part of this process, any necessary changes to the simulator are identified and implemented during the design change review process.

VII.2.D Operator Training Program

RESPONSE:

It is anticipated that the MUR power uprate will have no impact on the current operator training program. Operator training on the plant changes required to support the MUR power uprate will be completed prior to MUR power uprate implementation.

Training on operation and maintenance of the Cameron LEFM CheckPlus System, will be developed and completed prior to implementation of the MUR power uprate.

VII.3 A statement confirming licensee intent to complete the modifications identified in Item 2. above (including the training of operators), prior to implementation of the power uprate.

RESPONSE:

Changes/modifications to the simulator and the associated manuals and instructional materials will be implemented in accordance with the ONS engineering change process to capture all plant changes as a result of the MUR power uprate. Duke Energy will complete modifications related to the MUR power uprate and complete the training of operators prior to implementation of the MUR power uprate.

VII.4 A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

RESPONSE:

Operating Procedures (OPs) have been reviewed and required changes will be documented and implemented as part of the normal engineering change and licensing processes, as noted in Section VII.2.A, the procedure related to temporary operation above full steady-state licensed power levels will be reviewed and modified as necessary.

VII.5 A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

VII.5.A A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

RESPONSE:

VII.5.A.i Non-Radiological Effluents

Limits for pertinent non-radiological discharge to the environment are regulated via a National Pollution Discharge Elimination System (NPDES) Permit. All three units discharge through one structure near the Keowee dam. The NPDES Permit identifies both chemical and thermal discharge limits for the plant.

Chemical Discharge: The MUR power uprate will not change chemical discharges controlled by the NPDES permit. No changes in the types or amounts of effluents released into the environment will occur due to the uprate.

Thermal discharge: Thermal discharge will remain controlled administratively, as necessary to comply with the NPDES requirements. A review of current documentation indicates that NPDES requirements have been consistently met for the plants' NPDES permit.

Thermal discharge limits include:

Effluent Temperature Limit:

Daily Maximum - Discharge temperature shall not exceed 37.8°C (100°F) for a time period more than two hours, unless critical hydrological and meteorological conditions are combined with high customer demand, which cannot be met from other sources as determined by the System Operations Center. Under these latter conditions, the discharge temperature shall not be allowed to exceed 39.4°C (103°F).

Temperature Rise:

Daily Maximum - 12.2°C (22°F), when the intake temperature is greater than 20°C (68°F).

An assessment of the MUR power uprate, using the PEPSE thermal model and a maximum CCW system cooling water intake temperature of 85°F, predicts that an increase in power output of 2% would result in a Delta-T increase of about 0.3 °F. If the NPDES permit thermal limits were approached, administrative controls are in place to assure that power levels can be reduced to accommodate permit requirements. Therefore, the level of power and resulting temperature increase is not expected to result in circumstances that will cause NPDES Permit limits to be exceeded.

VII.5.A.ii Radiological Effluents:

During normal operation, the administrative control of release rate of radwaste systems does not change with operating power. Thus, no impact on routine licensed releases is anticipated. A review of historical liquid and gaseous data indicates that resultant doses are a very small fraction of annual limits. This data provides verification that the MUR power uprate will not cause doses from waste liquid and gaseous waste releases to exceed allowable limits.

A review of recent plant Annual Radiological Environmental Operating Reports showed that the impact as a result of plant releases is generally absent. Where present in the environment, radioactivity remains at a small fraction of allowable limits.

VII.5.B A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

RESPONSE:

Radiological dose has been evaluated relative to a proposed MUR power uprate for the ONS units. An increase in individual and cumulative occupational radiation exposure is not expected because the MUR power increase is bounded by the existing analyses of record at 102% of the current rated thermal power as discussed in Section II. Individual worker exposures will be maintained within limits by the station Radiation Protection and ALARA Programs. Thus, no impact on radiological dose is anticipated.

The ONS Technical Specifications and the Offsite Dose Calculation Manual (ODCM) implement the regulations that control off-site doses to the public. The ODCM, contains a methodology for conservatively assessing off-site doses on an ongoing basis. This assures that regulatory limits will not be exceeded and that appropriate actions can be implemented if ALARA dose objectives are approached. Dose evaluations for accident scenarios reported in Chapter 15 of the ONS UFSAR already take into account, as applicable, an operating level of 102% of the baseline plant power rating and are discussed in Section II. No changes in the ODCM program are planned as part of the MUR power uprate process and doses will continue to be controlled to existing limits.

VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

RESPONSE:

A review of each of the MUR power uprate system evaluations was completed to determine any impacts to the Fire Protection Program and the Safe Shutdown Analysis. The MUR power uprate does not change or modify the credited equipment necessary for post fire safe shutdown nor does it reroute essential cables or relocate essential components credited by the safe shutdown analysis. Installation of the LEFM components was reviewed under the administrative controls of the ONS design change process and found to not adversely impact safe shutdown. Additional building heat-up will be minimal such that currently credited fire protection manual actions will not be prevented from being accomplished by their required time. Damage control procedures have actions to open doors, bring in fans, or use other methods to cool the environment for more suitable working conditions and to ensure proper operation of safe shutdown equipment. No new operator actions for fire protection were identified.

The HPSW System is credited for supplying water to the fire headers as discussed in Section VI.1.C.iii. There are no new uses of fire protection water as a result of the MUR power uprate.

VII.6.B Containment Coatings Program

RESPONSE:

Containment coatings are discussed in UFSAR Section 6.2.1.6. The proposed MUR power uprate does not have any impact on the programmatic aspects of the Coatings Program. The UFSAR LOCA containment response analyses remain bounding for the MUR power uprate. There were no changes to the containment analyses that would require a change to the containment design pressure or

temperature. Since the containment design pressure and temperature limits were used to qualify the Service Level 1 containment coatings, and those limits are not changing, the Service Level 1 containment coatings remain qualified under MUR power uprate conditions. Therefore, the MUR power uprate is bounded by current analysis of record and no changes are required.

VII.6.C Maintenance Rule Program

RESPONSE:

10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance in Nuclear Power Plants requires monitoring the performance or condition of SSCs. The Maintenance Rule Program establishes a method to monitor system performance against criteria and acts to improve poor system performance. ONS has developed and has implemented the Maintenance Rule Program for applicable systems per these requirements. The proposed MUR power uprate does not have any impact on the programmatic aspects of the Maintenance Rule Program. It does not change any of the regulatory requirements of the program or in any way change the scope of the program. It does not add or delete any systems since the new LEFM screens out of the Maintenance Rule Program.

VII.6.D Motor- and Air-Operated Valve Programs

RESPONSE:

The programmatic requirements for the Motor-Operated Valve (MOV) program come from Generic Letter 89-10. These requirements include: (1) reviewing and documenting the design basis for each subject MOV; (2) determining the correct switch settings for each MOV; (3) setting the switches on each MOV to the correct settings; (4) testing each MOV under static and (if practicable) design basis conditions; (5) establishing procedures to maintain switch settings throughout the life of the plant including the effects of aging or degradation; (6) analyzing each switch failure and taking the proper corrective actions; (7) and as required by GL 96-05, instituting a program for periodic verification of the MOV switch settings. The proposed MUR power uprate does not have any impact on the programmatic aspects of the GL 89-10 program. It does not change any of the regulatory requirements or change the scope of the program.

The Air-Operated Valve (AOV) Program for ONS is not impacted by the MUR power uprate. The systems that contain AOVs within the program were evaluated and determined to continue to be within design parameters after implementation of the MUR power uprate. The required operating thrust/torque and actuator output capability for the AOVs are determined based on worst-case operating conditions within the licensing basis of the plant. These worst-case conditions, for which the AOVs are required to operate, remain unchanged due to the MUR power uprate. The MUR power uprate does not alter the basis, scope, or content of the AOV Program. No AOVs will be added or deleted from the program due to the MUR power uprate. No maintenance or material changes for any AOVs will be required.

VII.6.E Containment Leakage Rate Testing Program

RESPONSE:

The Containment Leakage Rate Testing Program is discussed in ONS Technical Specifications Section 5.5.2. The MUR power uprate does not have any impact on the programmatic aspects of the Appendix J Program. It does not change any of the regulatory requirements of the program or change the scope of the program. The MUR power uprate does not change containment peak pressure following a large break LOCA as discussed in Section II, above.

VIII Changes to technical specifications, protection system settings, and emergency system settings

VIII.1 A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

VIII.1.A A Description of the Change

RESPONSE:

The description of Technical Specification changes, including protection system settings, is provided in Section 2 of Enclosure 1. Revised Technical Specifications are attached, a marked up copy in Attachment 2 and a retyped copy in Attachment 3. Likewise, marked up Technical Specification Bases are provided in Attachment 4 and a retyped copy is provided as Attachment 5.

VIII.1.B Identification of Analyses Affected By and/or Supporting the Change

RESPONSE:

The calculations that support the MUR power uprate Technical Specification changes, or are affected by the MUR power uprate Technical Specification changes, are discussed below.

The ONS heat balance uncertainty calculation has been revised to calculate the uncertainty associated with the secondary heat balance after installation of the LEFMs. Site-specific calculations by Cameron of the accuracy of the installed LEFMs were used as input to the revised heat balance uncertainty analysis. These analyses are explained in Section I of this Enclosure.

The revised RPS overpower trip TS allowable values and the corresponding accident analyses that support the MUR power uprate conditions are discussed in Section II.

The Flux-Flow-Imbalance envelope was also reviewed based on the increased power level, and it was determined that no changes are required.

The calculation of the arming setpoint for ARTS was reviewed based on the power uprate. The arming setpoint was left at 50% power, which is a slight increase in megawatts. Because this system actuates based on pressure rather than power, this small change does not affect system operation.

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.

RESPONSE:

The justification for the Technical Specification changes is provided in Section 3 of Enclosure 1.

References for Section VIII:

None

ATTACHMENT 1 REGULATORY COMMITMENTS

The following table identifies those regulatory commitments planned to by Duke Energy Carolinas, LLC (Duke Energy) in this submittal. Other actions discussed in the submittal represent intended or planned actions by Duke Energy. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

	Action	Completion Date
1	Duke Energy will complete modifications related to the CRDM re-power as discussed in Enclosure 1, Section 1.1.	Prior to implementation of the MUR power uprate.
2	Duke Energy will complete modifications related to the SSF letdown line as discussed in Enclosure 1, Section 1.1.	Prior to implementation of the MUR power uprate.
3	A Selected Licensee Commitment will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is not functional. The SLC will include a requirement to perform channel calibration of the LEFM system pressure transmitters on a refueling outage frequency. This calibration ensures the measurement uncertainty of the outputs from these pressure transmitters remain bounded by the analysis and assumptions set forth in Cameron Engineering Uncertainty Analysis Reports.	Prior to implementation of the MUR power uprate.
4	The Cross Around Piping (CAP) between the high pressure and low pressure turbines will be re-rated.	Prior to implementation of the MUR power uprate
5	Duke Energy will complete modifications related to the MUR power uprate identified in Enclosure 2, VII.2.B.	Prior to implementation of the MUR power uprate.

ATTACHMENT 2 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION MARKUPS

2610

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2568~~ megawatts thermal.

B. Technical Specifications

XXX

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~411~~, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

XXX

A. <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2568~~ 2610 megawatts thermal.

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~413~~ YYY, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

- 3 -

A. Maximum Power Level

2610

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2568~~ megawatts thermal.

B. Technical Specifications

ZZZ

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~412~~, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

ZZZ

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt. Add superscript "*"
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CONTROL RODS verified fully inserted by two independent means, it is not necessary to account for a stuck CONTROL ROD in the SDM calculation. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.

XXX, YYY & ZZZ

*Following implementation of MUR on the respective unit, the value of RTP shall be 2610 MWt.

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

Add superscript "(f)"

Add superscript "(g)"

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower				
a. High Setpoint	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7(d)(e)	≤ 105.5% RTP with four pumps operating, and ≤ 80.5% RTP when reset for three pumps operating per LCO 3.4.4, "RCS Loops - MODES 1 and 2"
b. Low Setpoint	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 618°F
3. RCS High Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 1800 psig
5. RCS Variable Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR
6. Reactor Building High Pressure	1,2,3(c)	C	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.7	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR

XXX, YYY & ZZZ

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9.	Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 800 psig
10.	Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 75 psig
11.	Shutdown Bypass RCS High Pressure	$2^{(b)},3^{(b)}$ $4^{(b)},5^{(b)}$	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

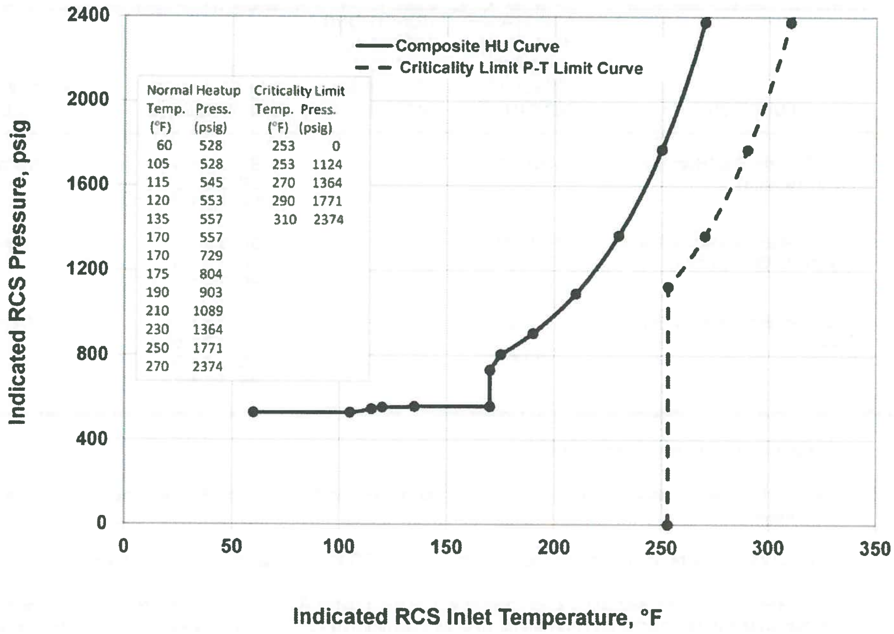
(d) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint or a value that is more conservative than the Nominal Trip Setpoint; otherwise the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodologies used to determine the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Selected Licensee Commitments Manual.

(f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in Selected Licensee Commitment 16.7.18, "Leading Edge Flow Meter."

(g) Following implementation of MUR on the respective unit, the value shall be 79.3% RTP.

XXX, YYY & ZZZ



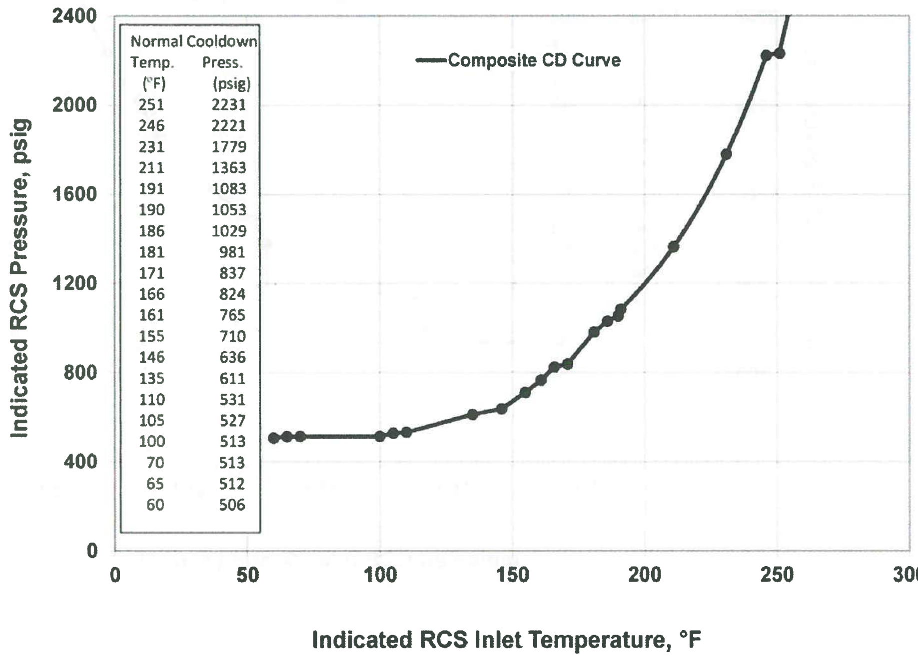
The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 1

44.6

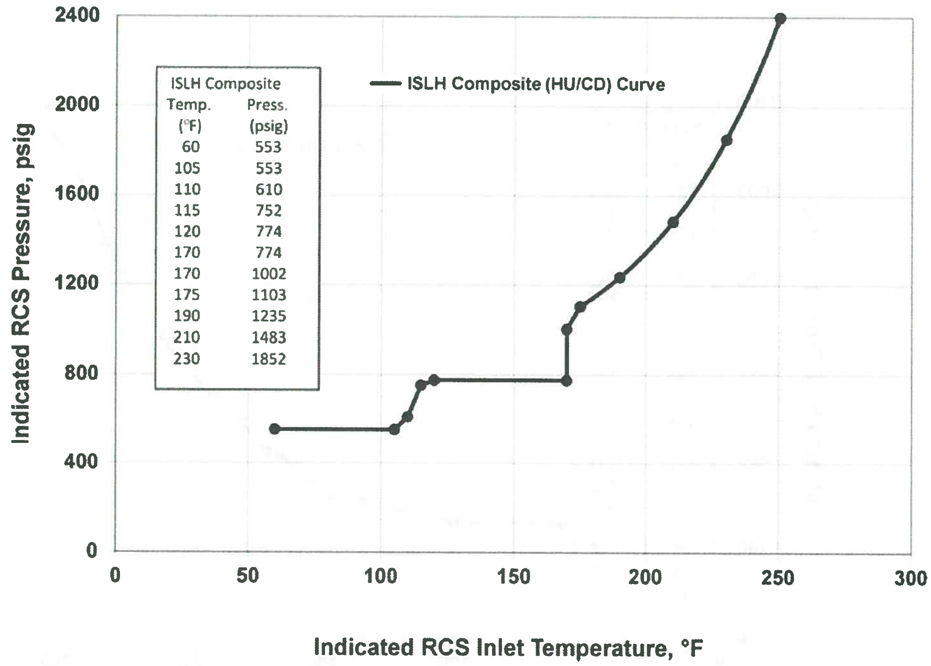


Normal Cooldown Temp. (°F)	Press. (psig)
251	2231
246	2221
231	1779
211	1363
191	1083
190	1053
186	1029
181	981
171	837
166	824
161	765
155	710
146	636
135	611
110	531
105	527
100	513
70	513
65	512
60	506

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First ~~54~~ 44.6 EFPY - Oconee Nuclear Station Unit 1

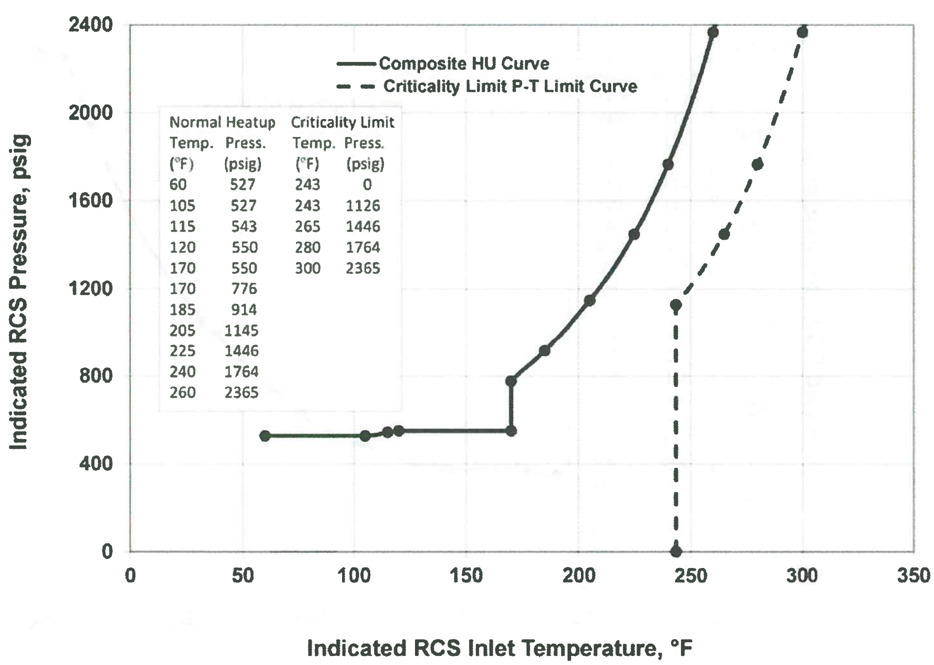


The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-3 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First ~~54~~ 44.6 EFPY - Oconee Nuclear Station Unit 1

RCS P/T Limits
3.4.3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

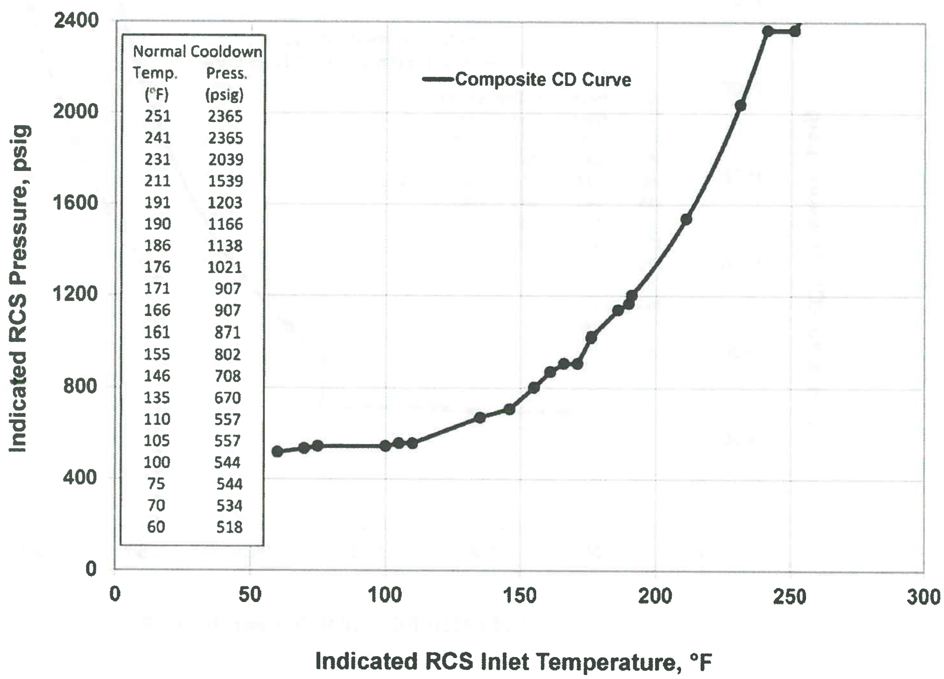
<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 2

45.3

RCS P/T Limits
3.4.3



Normal Cooldown Temp. (°F)	Press. (psig)
251	2365
241	2365
231	2039
211	1539
191	1203
190	1166
186	1138
176	1021
171	907
166	907
161	871
155	802
146	708
135	670
110	557
105	557
100	544
75	544
70	534
60	518

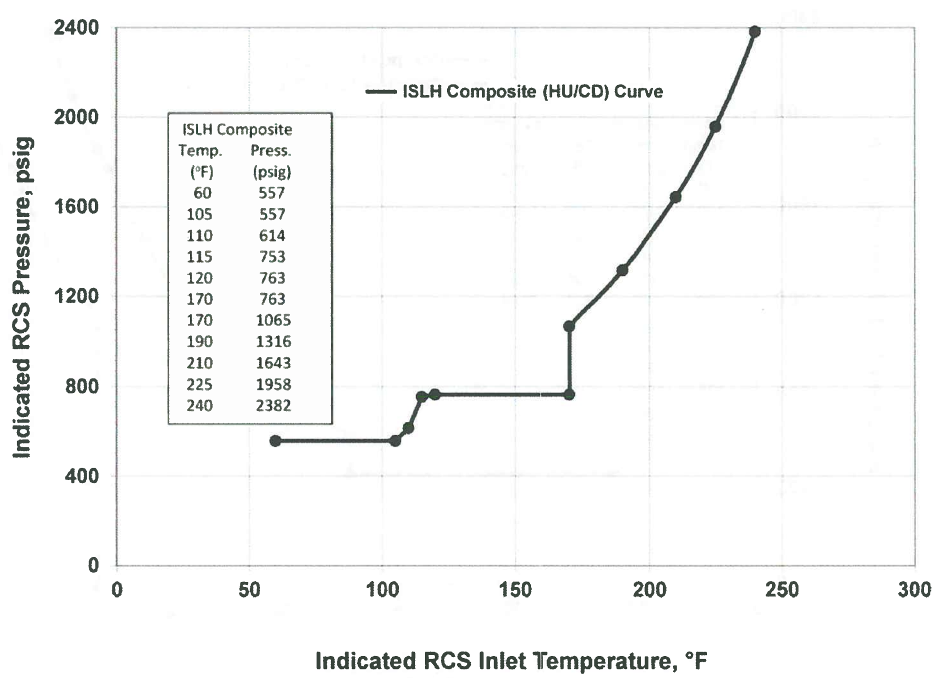
The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 2

45.3



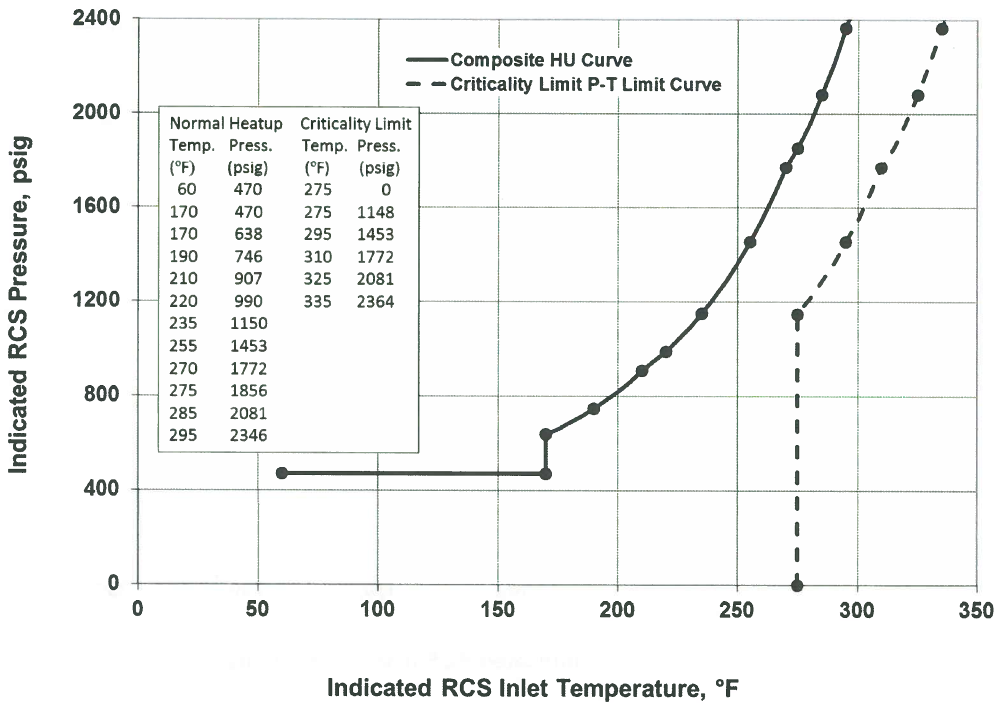
The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-6 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 2

45.3

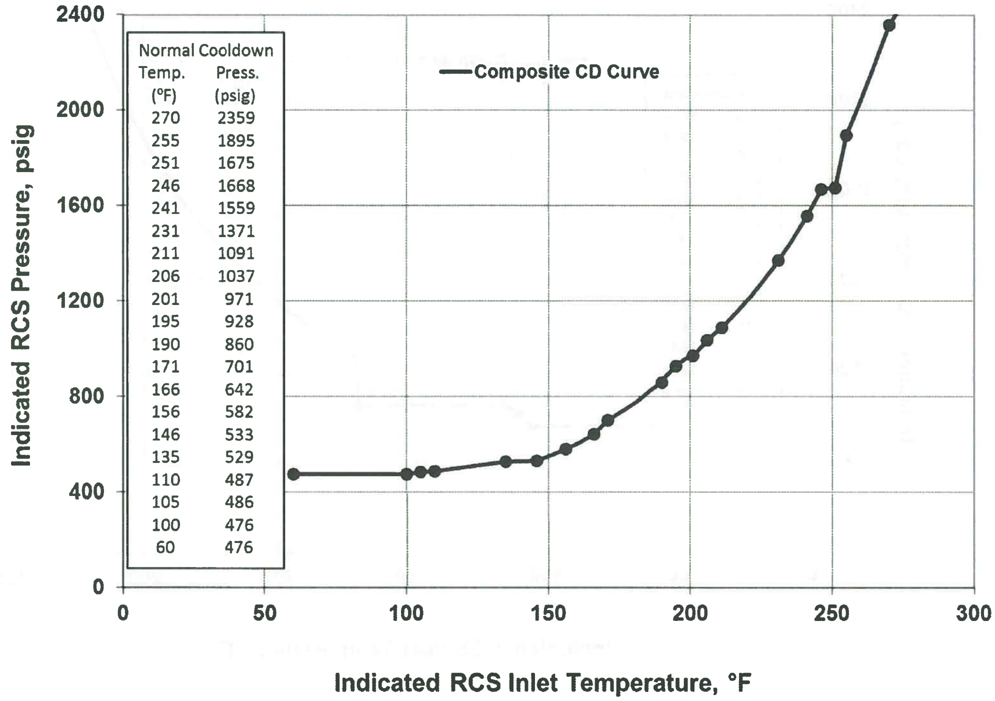


Normal Heatup Temp. (°F)	Normal Heatup Press. (psig)	Criticality Limit Temp. (°F)	Criticality Limit Press. (psig)
60	470	275	0
170	470	275	1148
170	638	295	1453
190	746	310	1772
210	907	325	2081
220	990	335	2364
235	1150		
255	1453		
270	1772		
275	1856		
285	2081		
295	2346		

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First ~~54~~ 43.8 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

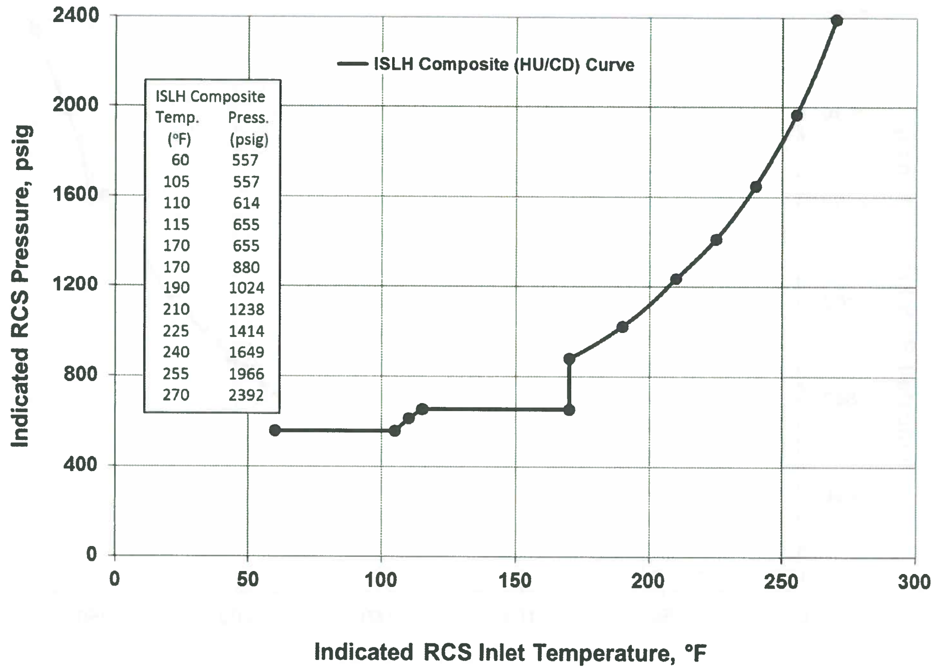
<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-8 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 3

43.8

RCS P/T Limits
3.4.3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

XXX, YYY & ZZZ

Figure 3.4.3-9 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First ~~54~~ EFPY - Oconee Nuclear Station Unit 3

43.8

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and:

1. THERMAL POWER is ≤ 75% RTP; and
2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.a (Nuclear Overpower – High Setpoint), Allowable Value of Table 3.3.1-1 is reset for 3 RCPs operating.

Add superscript "*"

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO 3.4.4.b not met.	A.1 Reset the RPS to satisfy the requirements of LCO 3.4.4.b.2.	10 hours
B. Required Action and associated Completion Time of Condition A not met. OR Requirements of LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	12 hours

* Following implementation of MUR on the respective unit, the value shall be 73.8% RTP.

XXX, YYY & ZZZ

ATTACHMENT 3 RETYPED FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. XXX, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1 . As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. YYY, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ZZZ, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT (QPT)	QPT shall be defined by the following equation and is expressed as a percentage. $QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt.*
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CONTROL RODS verified fully inserted by two independent means, it is not necessary to account for a stuck CONTROL ROD in the SDM calculation. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM; b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and c. There is no change in APSR position.

*Following implementation of MUR on the respective unit, the value of RTP shall be 2610 MWt.

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower				
a. High Setpoint	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7[(d)(e)]	≤ 105.5% RTP with four pumps operating, and ≤ 80.5% RTP[(g)] when reset for three pumps operating per LCO 3.4.4, "RCS Loops - MODES 1 and 2"[(f)]
b. Low Setpoint	2[(b)],3[(b)] 4[(b)],5[(b)]	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 618°F
3. RCS High Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[(c)]	C	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.7	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 800 psig
10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 75 psig
11. Shutdown Bypass RCS High Pressure	$2^{(b)}$,$3^{(b)}$ $4^{(b)}$,$5^{(b)}$	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

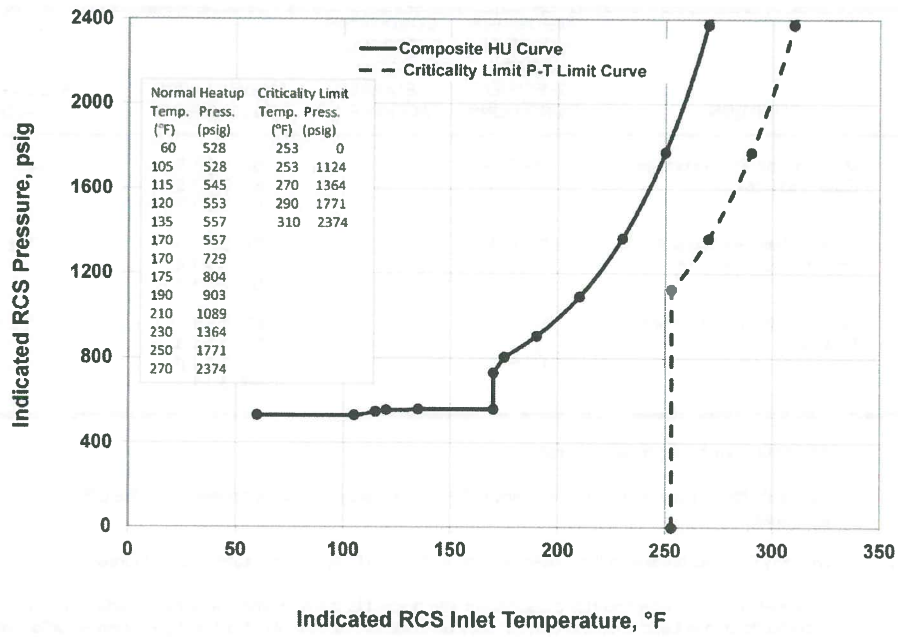
(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify 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 Nominal Trip Setpoint or a value that is more conservative than the Nominal Trip Setpoint; otherwise the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodologies used to determine the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Selected Licensee Commitments Manual.

(f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in Selected Licensee Commitment 16.7.18, "Leading Edge Flow Meter."

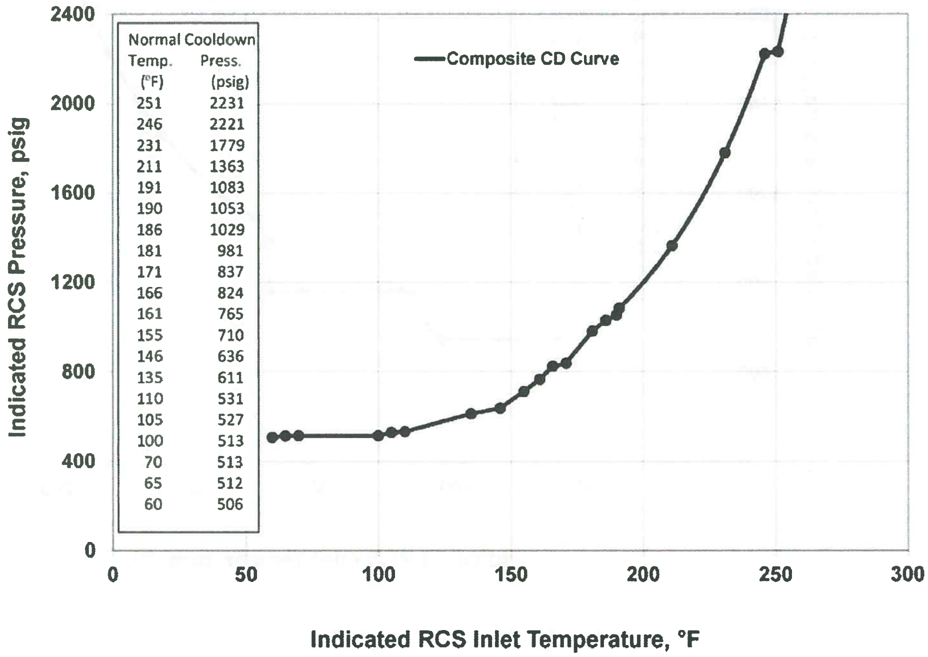
(g) Following implementation of MUR on the respective unit, the value shall be 79.3% RTP.



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1

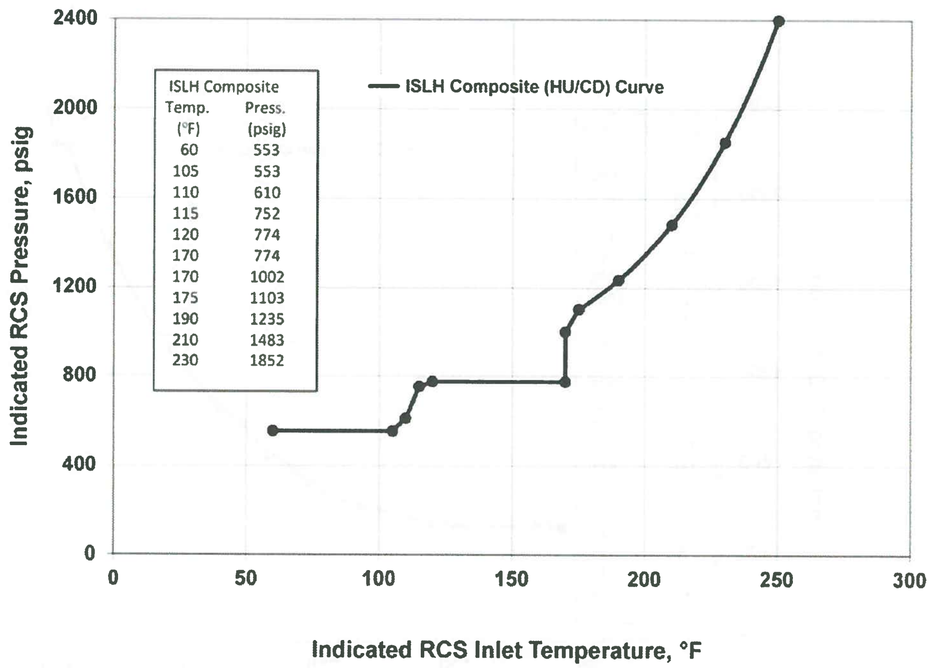


The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1

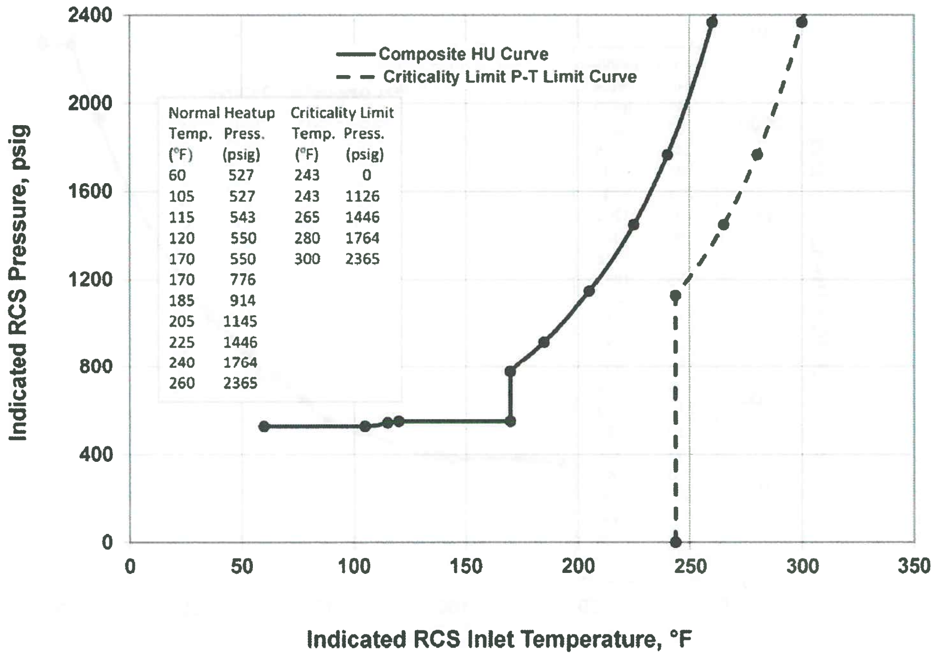
RCS P/T Limits
3.4.3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

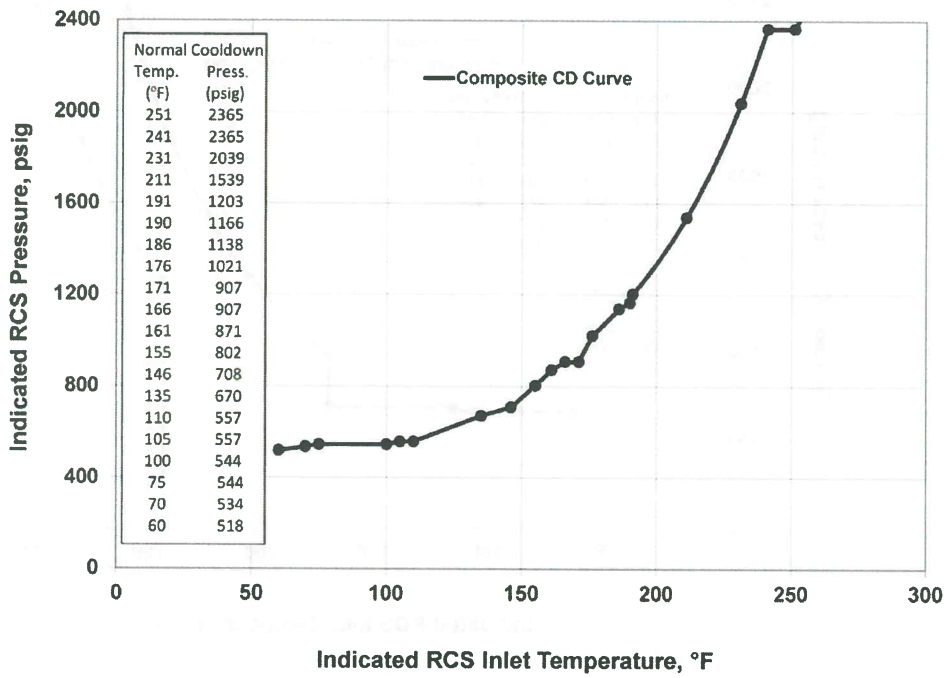
Figure 3.4.3-3 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

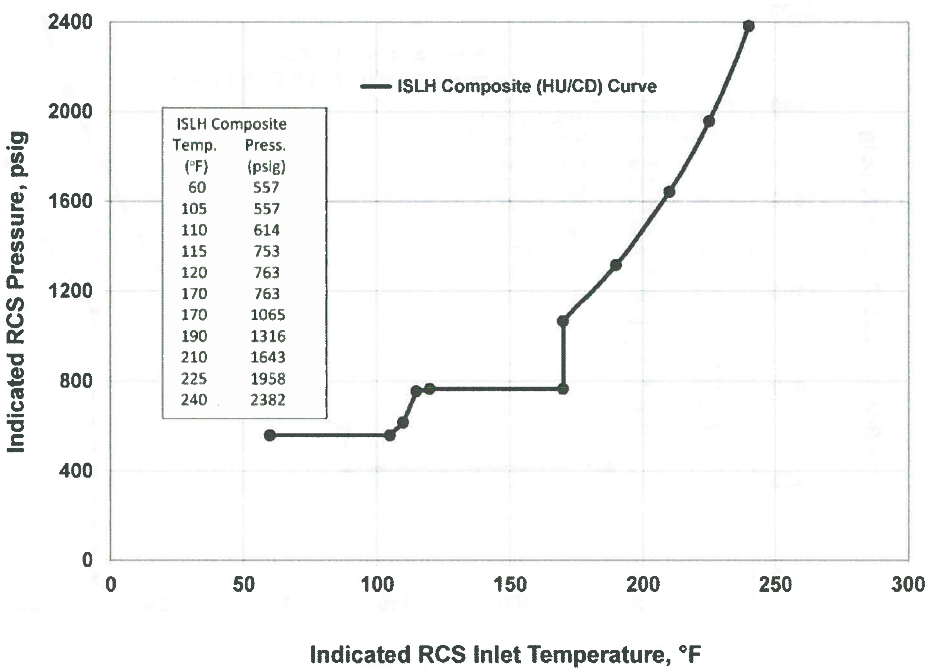
Figure 3.4.3-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

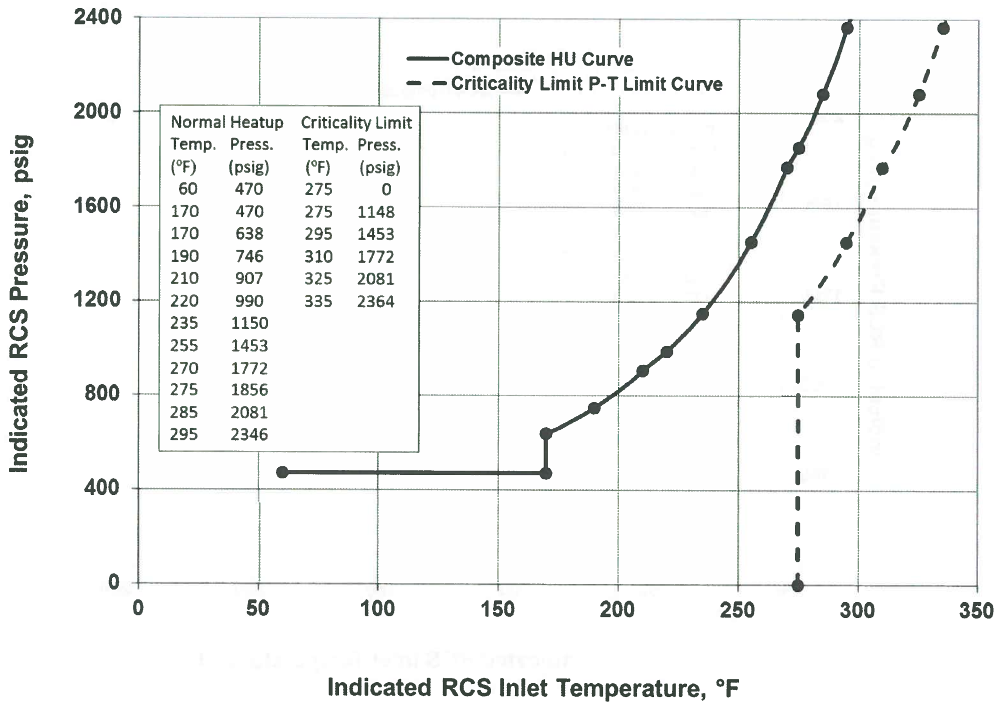
Figure 3.4.3-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-6 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2



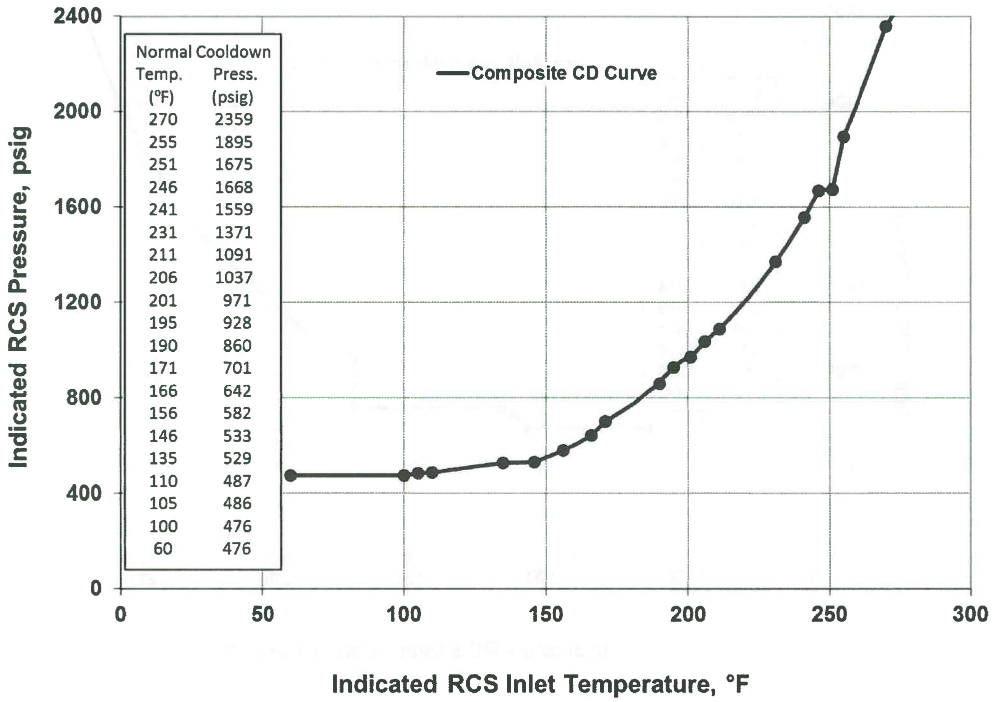
Normal Heatup Temp. (°F)	Normal Heatup Press. (psig)	Criticality Limit Temp. (°F)	Criticality Limit Press. (psig)
60	470	275	0
170	470	275	1148
170	638	295	1453
190	746	310	1772
210	907	325	2081
220	990	335	2364
235	1150		
255	1453		
270	1772		
275	1856		
285	2081		
295	2346		

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3

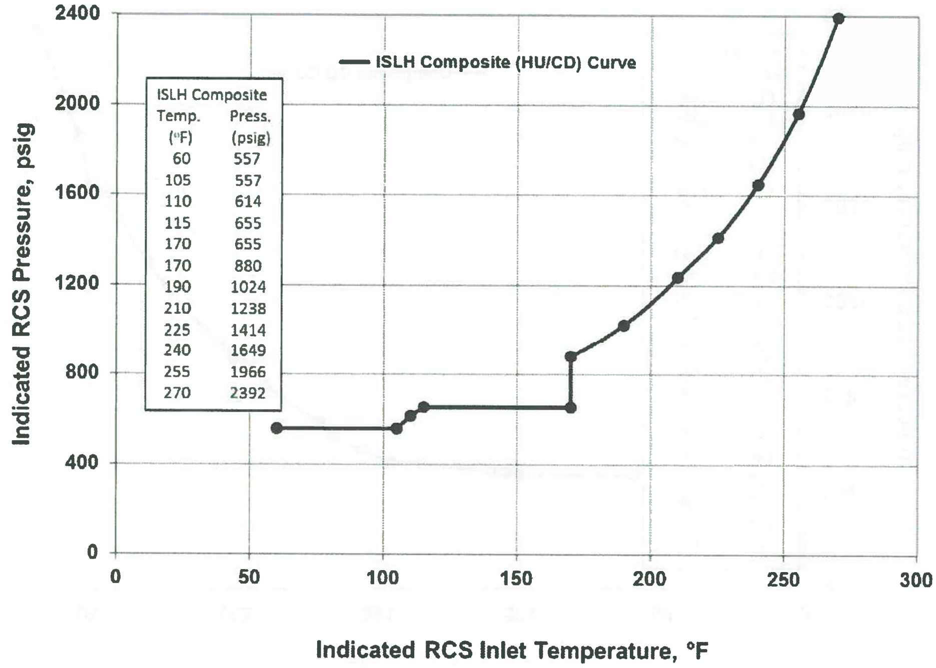
RCS P/T Limits
3.4.3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-8 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-9 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and:
   1. THERMAL POWER is ≤ 75% RTP*; and
   2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.a (Nuclear Overpower – High Setpoint), Allowable Value of Table 3.3.1-1 is reset for 3 RCPs operating.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO 3.4.4.b not met.	A.1 Reset the RPS to satisfy the requirements of LCO 3.4.4.b.2.	10 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Requirements of LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	12 hours

* Following implementation of MUR on the respective unit, the value shall be 73.8% RTP.

ATTACHMENT 4 MARKED UP TECHNICAL SPECIFICATIONS BASES

(for information only)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

operation, testing and subsequent calibration are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4.

For most RPS Functions, the Allowable Value in conjunction with the nominal trip setpoint ensure that the departure from nucleate boiling (DNB), center line fuel melt, or RCS pressure SLs are not challenged. Cycle specific values for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

a. Nuclear Overpower – High Setpoint

The Nuclear Overpower – High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core neutron leakage flux.

There is a setpoint for 4 and 3 RCP operation. The purpose of the 3 RCP trip is to provide protection for power excursion events initiated from 3 RCP operation, most notably the small steam line break.

For Unit(s) without the Measurement Uncertainty Recapture (MUR) power uprate complete, rated thermal power is 2568 MWt, and the heat balance accuracy of 2% means that rated power plus uncertainty is 2619 MWt. For Unit(s) with the MUR power uprate complete, rated thermal power is 2610 MWt, and the heat balance accuracy of 0.34% means that rated power plus uncertainty is still 2619 MWt.

For Unit(s) without the MUR power uprate complete, the nuclear overpower trip setpoint is 105.5% of 2568 MWt, or 2709 MWt. For Unit(s) with the MUR power uprate complete, the nuclear overpower trip setpoint is 105.5% of 2610 MWt, or 2754 MWt.

For Unit(s) with the MUR power uprate complete, the nuclear overpower trip setpoint with 3 RCPs operating is manually reduced to 79.3% of 2610 MWt.

The Nuclear Overpower – High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to prevent exceeding acceptable fuel damage limits.

Thus, the Nuclear Overpower – High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower Flux/Flow Imbalance, provide more direct protection. The role of the Nuclear Overpower – High Setpoint trip is to limit reactor THERMAL POWER below the highest

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the reactor coolant is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the reactor coolant include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

(The licensing analyses prior to the MUR power uprate were done for 75% of 2568 MWt = 1926 MWt. This equates to 73.8% of the post-MUR power level of 2610 MWt.)

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. During ~~With~~ three pumps ~~in~~ operation, the reactor power level is restricted to ~~75% RTP~~ 73.8% RTP (for Unit(s) with the Measurement Uncertainty Recapture (MUR) power uprate complete) and 75% RTP (for Unit(s) without the MUR power uprate complete) to ~~preserve the core power to flow relationship, thus maintaining the margin to DNB.~~ The intent of the specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

The Reactor Protection System (RPS) trip setpoint based on flux/flow/imbalance is automatically reduced when one pump is taken out of service; manual resetting is not necessary.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the accident analyses initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The analyses that are of most importance to RCP operation are the two pump coastdown, single pump locked rotor, and single pump broken shaft (Ref. 1).

Steady state DNB analysis has been performed for four, and three pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature protective limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level equal to the Nuclear Overpower – High Setpoint - 4 reactor coolant pumps running trip setpoint plus instrument uncertainty and conservatism. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the power to flow ratio of the RPS nuclear overpower trip setpoint based on flux/flow/imbalance and the Nuclear Overpower – High Setpoint – 3 reactor coolant pumps running once it has been reset by the operators. The maximum power level for three pump operation is ~~75%~~ 73.8% RTP for Unit(s) with the MUR power uprate complete and 75% RTP for Unit(s) Without the MUR power uprate complete. RTP and is based on the three pump flow as a fraction of the four pump flow at full power.

Continued power operation with two RCPs removed from service is not allowed by this Specification.

RCS Loops – MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. Flow is represented by the number of RCPs in operation in both RCS loops for removal of heat by the two SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if only three pumps are available, power must be reduced as must the Nuclear Overpower – High Setpoint – 3 reactor coolant pumps.

ATTACHMENT 5 RETYPED TECHNICAL SPECIFICATION BASES

(for information only)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

operation, testing and subsequent calibration are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4.

For most RPS Functions, the Allowable Value in conjunction with the nominal trip setpoint ensure that the departure from nucleate boiling (DNB), center line fuel melt, or RCS pressure SLs are not challenged. Cycle specific values for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

   a. Nuclear Overpower – High Setpoint

      The Nuclear Overpower – High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core neutron leakage flux.

      There is a setpoint for 4 and 3 RCP operation. The purpose of the 3 RCP trip is to provide protection for power excursion events initiated from 3 RCP operation, most notably the small steam line break.

      For Unit(s) without the Measurement Uncertainty Recapture (MUR) power uprate complete, rated thermal power is 2568 MWt, and the heat balance accuracy of 2% means that rated power plus uncertainty is 2619 MWt. For Unit(s) with the MUR power uprate complete, rated thermal power is 2610 MWt, and the heat balance accuracy of 0.34% means that rated power plus uncertainty is still 2619 MWt.

      For Unit(s) without the MUR power uprate complete, the nuclear overpower trip setpoint is 105.5% of 2568 MWt, or 2709 MWt. For Unit(s) with the MUR power uprate complete, the nuclear overpower trip setpoint is 105.5% of 2610 MWt, or 2754 MWt.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Nuclear Overpower – High Setpoint (continued)

For Unit(s) with the MUR power uprate complete, the nuclear overpower trip setpoint with 3 RCPs operating is manually reduced to 79.3% of 2610 MWt.

The Nuclear Overpower – High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to prevent exceeding acceptable fuel damage limits.

Thus, the Nuclear Overpower – High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower Flux/Flow Imbalance, provide more direct protection. The role of the Nuclear Overpower – High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower – High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident and the rod ejection accident. By providing a trip during these events, the Nuclear Overpower – High Setpoint trip protects the unit from excessive power levels and also serves to limit reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the reactor coolant is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the reactor coolant include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. During three pump operation, the reactor power level is restricted to 73.8% RTP (for Unit(s) with the Measurement Uncertainty Recapture (MUR) power uprate complete) and 75% RTP (for Unit(s) without the MUR complete) to preserve the core power to flow relationship, thus maintaining the margin to DNB. (The licensing analyses prior to the MUR power uprate were done for 75% of 2568 MWt = 1926 MWt. This equates to 73.8% of the post-MUR power level of 2610 MWt.) The intent of the specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Reactor Protection System (RPS) trip setpoint based on flux/flow/imbalance is automatically reduced when one pump is taken out of service; manual resetting is not necessary.

Safety analyses contain various assumptions for the accident analyses initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The analyses that are of most importance to RCP operation are the two pump coastdown, single pump locked rotor, and single pump broken shaft (Ref. 1).

Steady state DNB analysis has been performed for four, and three pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature protective limit (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level equal to the Nuclear Overpower – High Setpoint - 4 reactor coolant pumps running trip setpoint plus instrument uncertainty and conservatism. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the power to flow ratio of the RPS nuclear overpower trip setpoint based on flux/flow/imbalance and the Nuclear Overpower – High Setpoint – 3 reactor coolant pumps running once it has been reset by the operators. The maximum power level for three pump operation is 73.8% RTP for Unit(s) with the MUR power uprate complete and 75% RTP for (Unit(s) without the MUR complete and is based on the three pump flow as a fraction of the four pump flow at full power.

Continued power operation with two RCPs removed from service is not allowed by this Specification.

RCS Loops – MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2).

ATTACHMENT 6 NON-PROPRIETARY CAMERON REPORTS

Caldon® Ultrasonic Engineering Reports:

ER-813NP Rev 6, ER-824NP Rev 6, ER-825NP Rev 6, and ER-855NP Rev 0

sensia

Rockwell Automation + Schlumberger

Caldon® Ultrasonics

Engineering Report: ER-813NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 1 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Prepared by: Ryan Hannas
Reviewed by: Mike James MJ
Reviewed for Proprietary Information by: Joanna Phillips JMP

November 2019

Rockwell Automation + Schlumberger

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Engineering Report: ER-813NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 1 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Table of Contents

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1.0 INTRODUCTION

The LEFM✓ and LEFM✓+[1] are advanced ultrasonic systems that accurately determine the volume flow and temperature of feedwater in nuclear power plants. Using a feedwater pressure signal input to the LEFM✓ and LEFM✓+: mass flow can be determined and, along with the temperature output are used along with plant data to compute reactor core thermal power. The technology underlying the LEFM✓ ultrasonic instruments and the factors affecting their performance are described in a topical report, Reference 1, and a supplement to this topical report, Reference 2. The LEFM✓+, which is made of two LEFM✓ subsystems, is described in another supplement to the topical report, Reference 3. The exact amount of the uprate allowable under a revision to 10CFR50 Appendix K depends not only on the accuracy of the LEFM✓+ instrument, but also on the uncertainties in other inputs to the thermal power calculation.

It is the purpose of this document to provide an analysis of the uncertainty contribution of the LEFM✓+ System [ ][2] to the overall thermal power uncertainty of Oconee Unit 1 Nuclear Generating Station (Appendix B).

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The uncertainties in mass flow and feedwater temperature are also used in the calculation of the overall thermal power uncertainty (Appendix B). [ ] A detailed discussion of the methodology for combining these terms is described in Reference 3.

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This analysis is a bounding analysis for the Oconee Unit 1 Nuclear Generating Station. [ ]

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[ ]

[ ]

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2.0 SUMMARY

For Oconee Unit 1 Nuclear Generating Station, Revision 6 results are as follows:

1. The mass flow uncertainty approach is documented in Reference 3. The uncertainty in the LEFM✓+'s mass flow of feedwater is as follows:
    - Fully Functional LEFM✓+ system mass flow uncertainty is [ ]
    - Maintenance Mode LEFM✓+ system mass flow uncertainty is [ ].

   [ ]

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2. The uncertainty in the LEFM✓+ feedwater temperature is as follows:
    - Fully Functional LEFM✓+ system temperature uncertainty is [ ]
    - Maintenance Mode LEFM✓+ system the uncertainty is [ ]

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3. The total thermal power uncertainty approach is documented in Reference 3 and Appendix B of this document. The total uncertainty in the determination of thermal power uses the LEFM✓+ system parameters and plant specific parameters, i.e., heat gain/losses, etc. and is as follows:
    - Thermal power uncertainty using a Fully Functional LEFM✓+ system is [ ]
    - Thermal power uncertainty using a Maintenance Mode LEFM✓+ system is [ ]

   [ ]

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3.0 APPROACH

All errors and biases are calculated and combined according to the procedures defined in Reference 4 and Reference 5 in order to determine the 95% confidence and probability value. The approach to determine the uncertainty, consistent with determining set points, is to combine the random and bias terms by the means of the RSS approach provided that all the terms are independent, zero-centered and normally distributed.

Reference 4 defines the contributions of individual error elements through the use of sensitivity coefficients defined as follows:

A calculated variable P is determined by algorithm f, from measured variables X, Y, and Z.

$$P = f(X, Y, Z)$$

The error, or uncertainty in P, dP, is given by:

$$dP = \left.\frac{\partial f}{\partial X}\right|_{YZ} dX + \left.\frac{\partial f}{\partial Y}\right|_{XZ} dY + \left.\frac{\partial f}{\partial Z}\right|_{XY} dZ$$

As noted above, P is the determined variable--in this case, reactor power or mass flow-- which is calculated via measured variables X, Y, and Z using an algorithm f (X, Y, Z). The uncertainty or error in P, dP, is determined on a per unit basis as follows:

$$\frac{dP}{P} = \left\{\frac{X}{P}\left.\frac{\partial f}{\partial X}\right|_{YZ}\right\}\frac{dX}{X} + \left\{\frac{Y}{P}\left.\frac{\partial f}{\partial Y}\right|_{XZ}\right\}\frac{dY}{Y} + \left\{\frac{Z}{P}\left.\frac{\partial f}{\partial Z}\right|_{XY}\right\}\frac{dZ}{Z}$$

where the terms in brackets are referred to as the sensitivity coefficients.

If the errors or biases in individual elements (*dX/X, dY/Y, and dZ/Z* in the above equation) are all caused by a common (systematic) boundary condition (for example ambient temperature) the total error dP/P is found by summing the three terms in the above equation. If, as is more often the case, the errors in X, Y, and Z are independent of each other, then Reference 4 and 5 recommends and probability theory requires that the total uncertainty be determined by the root sum square as follows (for 95% confidence and probability):

$$\frac{dP}{P} = \sqrt{\left[\left(\left\{\frac{X}{P}\left.\frac{\partial f}{\partial X}\right|_{YZ}\right\}\frac{dX}{X}\right)^2 + \left(\left\{\frac{Y}{P}\left.\frac{\partial f}{\partial Y}\right|_{XZ}\right\}\frac{dY}{Y}\right)^2 + \left(\left\{\frac{Z}{P}\left.\frac{\partial f}{\partial Z}\right|_{XY}\right\}\frac{dZ}{Z}\right)^2\right]}$$

Obviously, if some errors in individual elements are caused by a combination of boundary conditions, some independent and some related (i.e., systematic) then a combination of the two procedures is appropriate.

4.0 OVERVIEW

The analyses that support the calculation of LEFM✓+ uncertainties are contained in the appendices to this document. The function of each appendix is outlined below.

Appendix A.1, LEFM✓+ Inputs

This appendix tabulates dimensional and other inputs to the LEFM✓+. The spreadsheet calculates other key dimensions and factors from these inputs (e.g., the face-to-face distance between pairs of transducer assemblies), which is used by the LEFM✓+ for the computation of mass flow and temperature.

Appendix A.2, LEFM✓+ Uncertainty Items/Calculations

This appendix calculates the uncertainties in mass flow and temperature as computed by the LEFM✓+ using the methodology described in Appendix E of Reference 1 and Appendix A of Reference 3[3], with uncertainties in the elements of these measurements bounded as described in both references[4]. Reference 6 provides a detailed comparison of the equations used in Appendix A of Reference 3 and this report. The spreadsheet calculations draw on the data of Appendix A.1 for dimensional information. It draws from Appendix A.4 [

] These uncertainties are an important factor in establishing the overall uncertainty of the LEFM✓+.

Trade Secret & Confidential Commercial Information

[3] Reference 3 (ER 157P-A) develops the uncertainties for the LEFM✓+ system. Because this system uses two measurement planes, the structure of its uncertainties differs somewhat that of an LEFM3.

[4] Reference 3 (ER 157P-A) revised some of the time measurement uncertainty bounds. The revised bounds are a conservative projection of actual performance of the LEFM hardware. ER 80P used bounds that were based on a conservative projection of theoretical performance.

Appendix A.3, Meter Factor Calibration and Accuracy Assessment

The calibration test report for the spool piece(s) establishes the overall uncertainty in the profile factor of the LEFM3+. [

]

Appendix A.4, [ ]

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Appendix A.5, [ ]

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Appendix B, Total Thermal Power and Mass Flow Uncertainties using the LEFM ✓+ System

The total thermal power uncertainty due to the LEFM✓+ is calculated in this appendix, using the results of Appendix A.2, A.4 and A.5. Plant supplied steam conditions (which enter into the computation of errors due to feedwater temperature) are used for this computation. This appendix also computes the fraction of the uncertainty in feedwater temperature that is systematically related to the mass flow uncertainty.

5.0 REFERENCES

1) Cameron Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System", Rev. 0.

2) Cameron Engineering Report ER-160P, "Supplement to Topical Report ER 80P: Basis for a Power Uprate with the LEFM System", May 2000.

3) Cameron Engineering Report ER-157(P-A), "Supplement to Cameron Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus", dated May 2008, Revision 8 and Revision 8 Errata.

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7) ER-855 Rev.0, "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3", dated September 2010.

Appendix A

Appendix A.1, LEFM✓+ Inputs

Appendix A.2, LEFM✓+ Uncertainty Items/Calculations

Appendix A.3, Meter Factor Calibration and Accuracy Assessment

Appendix A.4, [ ]

Appendix A.5, [ ]

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Appendix A.1

LEFM✓+ Inputs

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.2

LEFM✓+ Uncertainty Items/Calculations

No attachment to follow, as Appendix is Proprietary in its Entirety

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Rockwell Automation + Schlumberger

Appendix A.3

LEFM✓+ Spool Piece(s) Meter Factor Calculation and Accuracy Assessment

Reference Cameron Engineering Report ER-855 Rev.0, "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3", dated September 2010

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Rockwell Automation + Schlumberger

Appendix A.4

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No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.5

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No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix B

Total Thermal Power and Mass Flow Uncertainty using the LEFM✓+ System

No attachment to follow, as Appendix is Proprietary in its Entirety

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Rockwell Automation + Schlumberger

Caldon® Ultrasonics

Engineering Report: ER-824NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 2 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Prepared by: Ryan Hannas
Reviewed by: Jonathan Lent
Reviewed for Proprietary Information by: Joanna Phillips

November 2019

Rockwell Automation + Schlumberger

Printed in the United States of America.

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Sensia
Rockwell Automation + Schlumberger

Engineering Report: ER-824NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 2 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Table of Contents

Trade Secret & Confidential Commercial Information

1.0 INTRODUCTION

The LEFM✓ and LEFM✓+[1] are advanced ultrasonic systems that accurately determine the volume flow and temperature of feedwater in nuclear power plants. Using a feedwater pressure signal input to the LEFM✓ and LEFM✓+: mass flow can be determined and, along with the temperature output are used along with plant data to compute reactor core thermal power. The technology underlying the LEFM✓ ultrasonic instruments and the factors affecting their performance are described in a topical report, Reference 1, and a supplement to this topical report, Reference 2. The LEFM✓+, which is made of two LEFM✓ subsystems, is described in another supplement to the topical report, Reference 3. The exact amount of the uprate allowable under a revision to 10CFR50 Appendix K depends not only on the accuracy of the LEFM✓+ instrument, but also on the uncertainties in other inputs to the thermal power calculation.

It is the purpose of this document to provide an analysis of the uncertainty contribution of the LEFM✓+ System [ ][2] to the overall thermal power uncertainty of Oconee Unit 2 Nuclear Generating Station (Appendix B).

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The uncertainties in mass flow and feedwater temperature are also used in the calculation of the overall thermal power uncertainty (Appendix B). [ ] A detailed discussion of the methodology for combining these terms is described in Reference 3.

Trade Secret & Confidentia Commercia Information

This analysis is a bounding analysis for the Oconee Unit 2 Nuclear Generating Station. [ ]

Trade Secret & Confidentia Commercia Information

[1] [ ]

[2] [ ]

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2.0 SUMMARY

For Oconee Unit 2 Nuclear Generating Station, Revision 6 results are as follows:

1. The mass flow uncertainty approach is documented in Reference 3. The uncertainty in the LEFM✓+'s mass flow of feedwater is as follows:
   - Fully Functional LEFM✓+ system mass flow uncertainty is [ ]
   - Maintenance Mode LEFM✓+ system mass flow uncertainty is [ ]

   [ ]

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2. The uncertainty in the LEFM✓+ feedwater temperature is as follows:
   - Fully Functional LEFM✓+ system temperature uncertainty is [ ]
   - Maintenance Mode LEFM✓+ system the uncertainty is [ ]

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3. The total thermal power uncertainty approach is documented in Reference 3 and Appendix B of this document. The total uncertainty in the determination of thermal power uses the LEFM✓+ system parameters and plant specific parameters, i.e., heat gain/losses, etc. and is as follows:
   - Thermal power uncertainty using a Fully Functional LEFM✓+ system is [ ]
   - Thermal power uncertainty using a Maintenance Mode LEFM✓+ system is [ ]

   [ ]

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3.0 APPROACH

All errors and biases are calculated and combined according to the procedures defined in Reference 4 and Reference 5 in order to determine the 95% confidence and probability value. The approach to determine the uncertainty, consistent with determining set points, is to combine the random and bias terms by the means of the RSS approach provided that all the terms are independent, zero-centered and normally distributed.

Reference 4 defines the contributions of individual error elements through the use of sensitivity coefficients defined as follows:

A calculated variable P is determined by algorithm f, from measured variables X, Y, and Z.

P = f (X, Y, Z)

The error, or uncertainty in P, dP, is given by:

$$dP = \left.\frac{\partial f}{\partial X}\right|_{YZ} dX + \left.\frac{\partial f}{\partial Y}\right|_{XZ} dY + \left.\frac{\partial f}{\partial Z}\right|_{XY} dZ$$

As noted above, P is the determined variable--in this case, reactor power or mass flow-- which is calculated via measured variables X, Y, and Z using an algorithm f (X, Y, Z). The uncertainty or error in P, dP, is determined on a per unit basis as follows:

$$\frac{dP}{P} = \left\{\frac{X}{P}\left.\frac{\partial f}{\partial X}\right|_{YZ}\right\}\frac{dX}{X} + \left\{\frac{Y}{P}\left.\frac{\partial f}{\partial Y}\right|_{XZ}\right\}\frac{dY}{Y} + \left\{\frac{Z}{P}\left.\frac{\partial f}{\partial Z}\right|_{XY}\right\}\frac{dZ}{Z}$$

where the terms in brackets are referred to as the sensitivity coefficients.

If the errors or biases in individual elements (*dX/X, dY/Y, and dZ/Z* in the above equation) are all caused by a common (systematic) boundary condition (for example ambient temperature) the total error dP/P is found by summing the three terms in the above equation. If, as is more often the case, the errors in X, Y, and Z are independent of each other, then Reference 4 and 5 recommends and probability theory requires that the total uncertainty be determined by the root sum square as follows (for 95% confidence and probability):

$$\frac{dP}{P} = \sqrt{\left[\left(\left\{\frac{X}{P}\left.\frac{\partial f}{\partial X}\right|_{YZ}\right\}\frac{dX}{X}\right)^2 + \left(\left\{\frac{Y}{P}\left.\frac{\partial f}{\partial Y}\right|_{XZ}\right\}\frac{dY}{Y}\right)^2 + \left(\left\{\frac{Z}{P}\left.\frac{\partial f}{\partial Z}\right|_{XY}\frac{dZ}{Z}\right)^2\right]}$$

Obviously, if some errors in individual elements are caused by a combination of boundary conditions, some independent and some related (i.e., systematic) then a combination of the two procedures is appropriate.

4.0 OVERVIEW

The analyses that support the calculation of LEFM✓+ uncertainties are contained in the appendices to this document. The function of each appendix is outlined below.

Appendix A.1, LEFM✓+ Inputs

This appendix tabulates dimensional and other inputs to the LEFM✓+. The spreadsheet calculates other key dimensions and factors from these inputs (e.g., the face-to-face distance between pairs of transducer assemblies), which is used by the LEFM✓+ for the computation of mass flow and temperature.

Appendix A.2, LEFM✓+ Uncertainty Items/Calculations

This appendix calculates the uncertainties in mass flow and temperature as computed by the LEFM✓+ using the methodology described in Appendix E of Reference 1 and Appendix A of Reference 3[3], with uncertainties in the elements of these measurements bounded as described in both references[4]. Reference 6 provides a detailed comparison of the equations used in Appendix A of Reference 3 and this report. The spreadsheet calculations draw on the data of Appendix A.1 for dimensional information. It draws from Appendix A.4 [ ] These uncertainties are an important factor in establishing the overall uncertainty of the LEFM✓+.

Trade Secret & Confidentia Commercia Information

[3] Reference 3 (ER 157P-A) develops the uncertainties for the LEFM✓+ system. Because this system uses two measurement planes, the structure of its uncertainties differs somewhat that of an LEFM3.

[4] Reference 3 (ER 157P-A) revised some of the time measurement uncertainty bounds. The revised bounds are a conservative projection of actual performance of the LEFM hardware. ER 80P used bounds that were based on a conservative projection of theoretical performance.

Appendix A.3, Meter Factor Calibration and Accuracy Assessment

The calibration test report for the spool piece(s) establishes the overall uncertainty in the profile factor of the LEFM3+. [

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Appendix A.4, [ ]

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Appendix A.5, [ ]

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Appendix B, Total Thermal Power and Mass Flow Uncertainties using the LEFM ✓+ System

The total thermal power uncertainty due to the LEFM✓+ is calculated in this appendix, using the results of Appendix A.2, A.4 and A.5. Plant supplied steam conditions (which enter into the computation of errors due to feedwater temperature) are used for this computation. This appendix also computes the fraction of the uncertainty in feedwater temperature that is systematically related to the mass flow uncertainty.

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7) ER-855 Rev.0, "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3", dated September 2010.

Appendix A

Appendix A.1, LEFM✓+ Inputs

Appendix A.2, LEFM✓+ Uncertainty Items/Calculations

Appendix A.3, Meter Factor Calibration and Accuracy Assessment

Appendix A.4, [ ]

Appendix A.5, [ ]

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Appendix A.1

LEFM✓+ Inputs

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.2

LEFM✓+ Uncertainty Items/Calculations

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.3

LEFM✓+ Spool Piece(s) Meter Calibration and Accuracy Assessment

Reference Cameron Engineering Report ER-855 Rev.0, "LEFM✓+ Meter Factor Calculation and Accuracy Assessment for Oconee Units 1, 2, and 3 Nuclear Generating Stations", September 2010

Appendix A.4

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Appendix A.5

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Appendix B

Total Thermal Power and Mass Flow Uncertainty using the LEFM✓+ System

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Engineering Report: ER-825NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 3 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Prepared by: Ryan Hannas
Reviewed by: Jonathan Lent
Reviewed for Proprietary Information by: Joanna Phillips

November 2019

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Rockwell Automation + Schlumberger

Printed in the United States of America.

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Sensia

Rockwell Automation + Schlumberger

Engineering Report: ER-825NP Rev 6

BOUNDING UNCERTAINTY ANALYSIS FOR THERMAL POWER DETERMINATION AT OCONEE UNIT 3 NUCLEAR GENERATING STATION USING THE LEFM✓+ SYSTEM

Table of Contents

Trade Secret & Confidential Commercial Information

1.0 INTRODUCTION

The LEFM✓ and LEFM✓+[1] are advanced ultrasonic systems that accurately determine the volume flow and temperature of feedwater in nuclear power plants. Using a feedwater pressure signal input to the LEFM✓ and LEFM✓+: mass flow can be determined and, along with the temperature output are used along with plant data to compute reactor core thermal power. The technology underlying the LEFM✓ ultrasonic instruments and the factors affecting their performance are described in a topical report, Reference 1, and a supplement to this topical report, Reference 2. The LEFM✓+, which is made of two LEFM✓ subsystems, is described in another supplement to the topical report, Reference 3. The exact amount of the uprate allowable under a revision to 10CFR50 Appendix K depends not only on the accuracy of the LEFM✓+ instrument, but also on the uncertainties in other inputs to the thermal power calculation.

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Trade Secret & Confidentia Commercia Information

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Trade Secret & Confidentia Commercia Information

This analysis is a bounding analysis for the Oconee Unit 3 Nuclear Generating Station. [ ]

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1 [ ]

2 [ ]

Trade Secret & Confidentia Commercia Information

2.0 SUMMARY

For Oconee Unit 3 Nuclear Generating Station, Revision 6 results are as follows:

1. The mass flow uncertainty approach is documented in Reference 3. The uncertainty in the LEFM✓+'s mass flow of feedwater is as follows:
   - Fully Functional LEFM✓+ system mass flow uncertainty is [ ]
   - Maintenance Mode LEFM✓+ system mass flow uncertainty is [ ].

   [ ]

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Trade Secret & Confidentia Commercia Information

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   - Thermal power uncertainty using a Fully Functional LEFM✓+ system is [ ]
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Trade Secret & Confidentia Commercia Information

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All errors and biases are calculated and combined according to the procedures defined in Reference 4 and Reference 5 in order to determine the 95% confidence and probability value. The approach to determine the uncertainty, consistent with determining set points, is to combine the random and bias terms by the means of the RSS approach provided that all the terms are independent, zero-centered and normally distributed.

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] These uncertainties are an important factor in establishing the overall uncertainty of the LEFM✓+.

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[3] Reference 3 (ER 157P-A) develops the uncertainties for the LEFM✓+ system. Because this system uses two measurement planes, the structure of its uncertainties differs somewhat that of an LEFM3.

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Appendix A.3, Meter Factor Calibration and Accuracy Assessment

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Trade Secret & Confidentia Commercia Information

Appendix A.4, [ ]

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Appendix A.5, [ ]

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The total thermal power uncertainty due to the LEFM✓+ is calculated in this appendix, using the results of Appendix A.2, A.4 and A.5. Plant supplied steam conditions (which enter into the computation of errors due to feedwater temperature) are used for this computation. This appendix also computes the fraction of the uncertainty in feedwater temperature that is systematically related to the mass flow uncertainty.

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7) ER-855 Rev.0, "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3", dated September 2010.

Appendix A

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Appendix A.2, LEFM✓+ Uncertainty Items/Calculations

Appendix A.3, Meter Factor Calibration and Accuracy Assessment

Appendix A.4, [ ]

Appendix A.5, [ ]

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Appendix A.1

LEFM✓+ Inputs

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.2

LEFM✓+ Uncertainty Items/Calculations

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix A.3

LEFM✓+ Spool Piece(s) Meter Factor Calculation and Accuracy Assessment

Reference Cameron Engineering Report ER-855 Rev.0, "LEFM✓+ Meter Factor Calculation and Accuracy Assessment for Oconee Units 1, 2, and 3 Nuclear Generating Stations", September 2010

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Appendix A.4

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Appendix A.5

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Appendix B

Total Thermal Power and Mass Flow Uncertainty using the LEFM✓+ System

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Caldon® Ultrasonics

Engineering Report: ER-855NP Rev 0

METER FACTOR CALCULATION AND ACCURACY ASSESSMENT FOR THE LEFM CHECK PLUS METERS AT OCONEE UNITS 1, 2 AND 3

Prepared by: John Spadaro
Reviewed by: Ryan Hannas RH
Reviewed for Proprietary Information by: Joanna Phillips JMV

November 2019

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Rockwell Automation + Schlumberger

Printed in the United States of America

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November 2019

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ER-855NP Rev 0, Meter Factor Calculation and Accuracy Assessment for the LEFMCheckPlus Meters at Oconee Units 1, 2 and 3

1.0 INTRODUCTION

1.1 Scope

This report documents calibration and uncertainty analysis for the following meters:

- Oconee Unit 1 flow elements S/N 38643 (Loop A) and S/N 38644 (Loop B)
- Oconee Unit 2 flow elements S/N 38648 (Loop A) and S/N 38645 (Loop B)
- Oconee Unit 3 flow elements S/N 38646 (Loop A) and S/N 38647 (Loop B)

This report includes:

- Meter factor(s) (e.g., profile factor) and meter factor uncertainty
- Description of the hydraulic models
- Description of the tests conducted
- [ ]

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1.2 Background

The flow meter measures the fluid velocity projected onto acoustic paths to determine the volumetric and mass flow. The meter makes the transit time measurements of ultrasonic energy pulses that travel between transducers and combines these with the distance separating the transducers to calculate the velocity. The flow meter uses eight acoustic paths that are arranged as two crossing planes, each plane containing four chords (essentially two four path meters). The meter calculates volumetric flow by numerically integrating the fluid-velocity chord length product along the chords.

[ ]

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The calibration test was performed at Alden Research Laboratories (Alden), an independent hydraulic laboratory. Alden provides flow rates up to ~4500 m^3/hr (~20,000 gpm). For these hydraulic models, the piping pressure losses and cavitation limited the calibration flow rate to ~3,400 m^3/hr (~15,000 gpm).

During the calibration, reference flow rates were determined by Alden using a weigh tank, fill times, fluid temperature and barometric pressure measurements. All elements of the lab measurements including weigh tank scale, time measurements, thermometers and pressure gages, are traceable to NIST standards. In order to determine the meter factor, the flow meter outputs are compared against the reference flow rates.

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The Oconee calibration test procedure was ALD-1134 Rev 0, which provided overall guidance for the test setup and test scope.

1.3 Report Summary

a. Table 1 provides the Oconee meter factors and uncertainties when calibrated in the installation model. Refer to Section 4 for a detailed summary.

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Table 1: Calibration Summary

b. [

]

c. Table 2 documents [

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[1] [

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Table 2: [ ]

[ ]

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2.0 CALIBRATION TESTS

The objectives for the calibration tests were to:

- Determine the meter factor in a piping configuration that models the installation site,
- Determine the sensitivity of the meter factor to variations in the hydraulic model,
- Determine the [ ]

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2.1 Meter Setup

2.1.1 Setup

The meter was installed in accordance with portions of Cameron Engineering Field Procedure EFP-61. Specifically, the portions of EFP-61 accomplished:

- [ ]
- [ ]
- [ ]

[ ]

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2.2 Installation Site Model

The hydraulic model configuration was designed as a hydraulic duplicate of the principle hydraulic features of the installation site (see Reference 1 for details). The model piping arrangements are shown in Figures 1 and 2 (for Unit 1 and Units 2/3, respectively). Although the drawings show two meters, only one meter was tested during each test series. The piping to the second meter was blocked immediately downstream of the second elbow. Figures 3 and 4 are photographs of the Alden Laboratories model for Unit 1 Loop A and Unit 3 Loop A. Figure 5 is a view of the eccentric orifice with 25% obstruction.

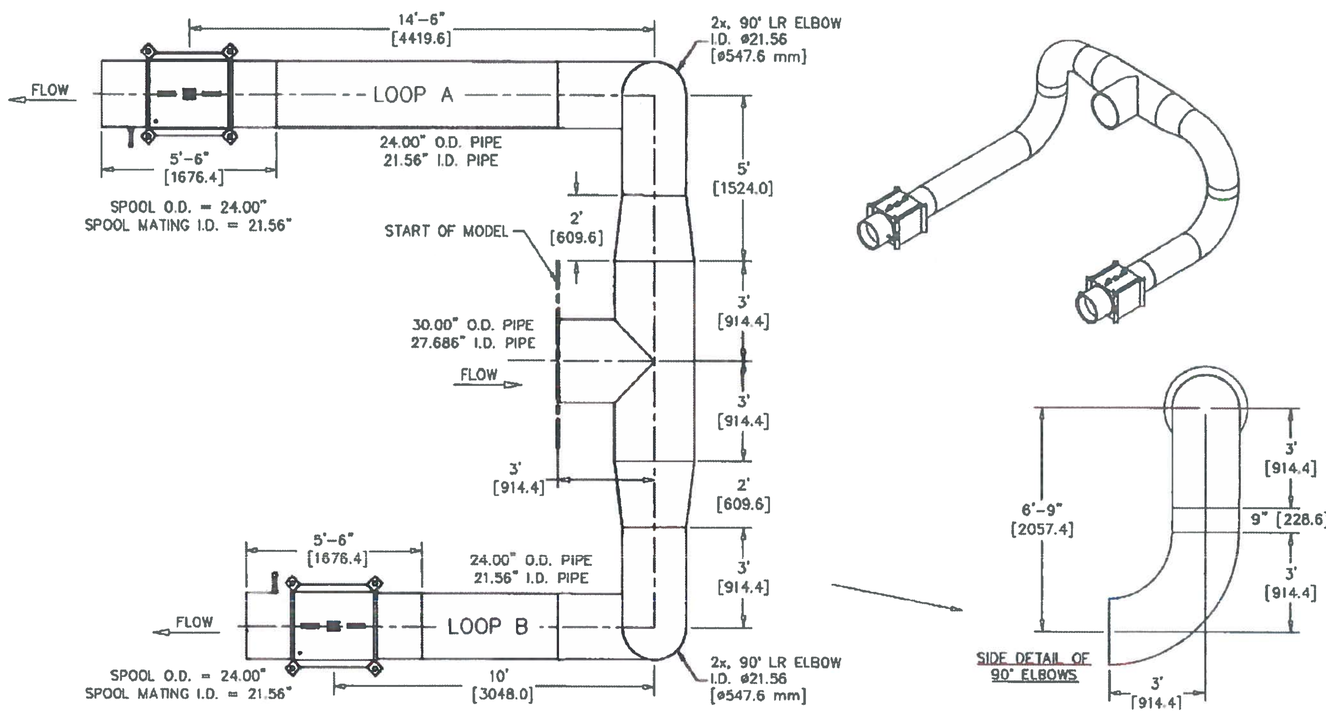

Figure 1: Unit 1 Hydraulic Models, Piping Lengths are +/- 2 inches/50 mm

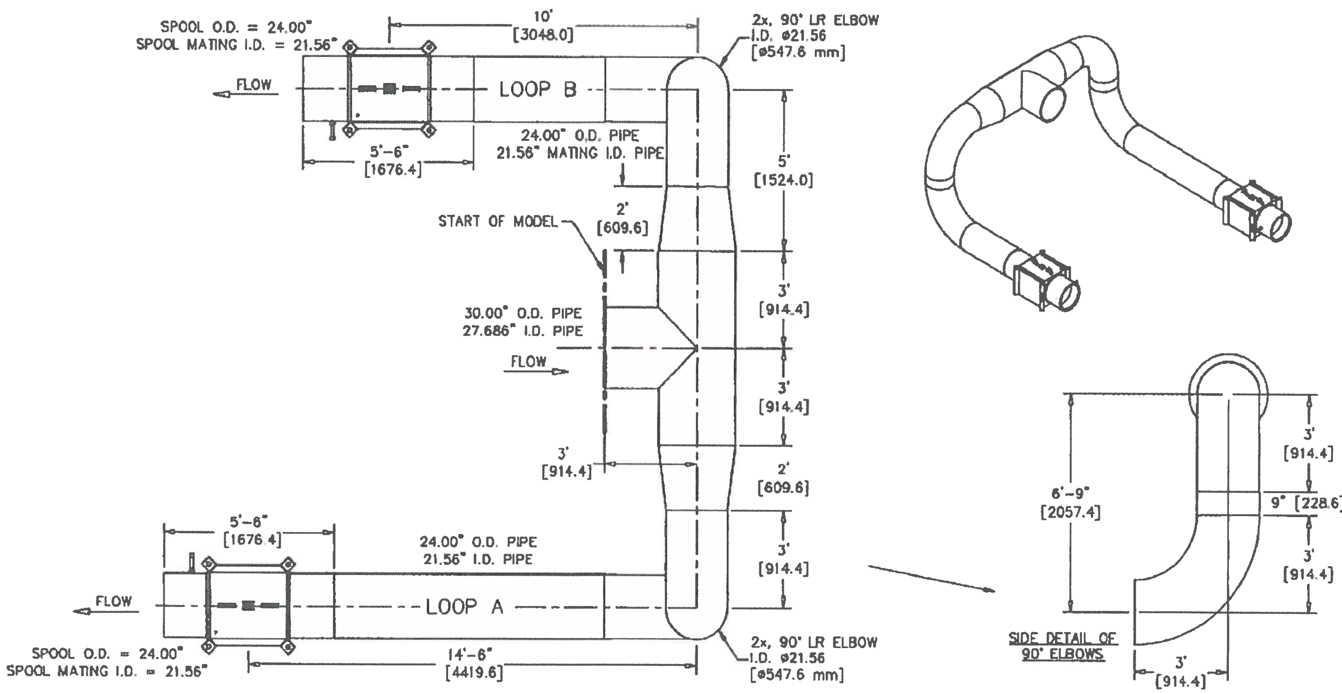

Figure 2: Units 2 and 3 Hydraulic Model, Piping Lengths are +/- 2 inches/50 mm

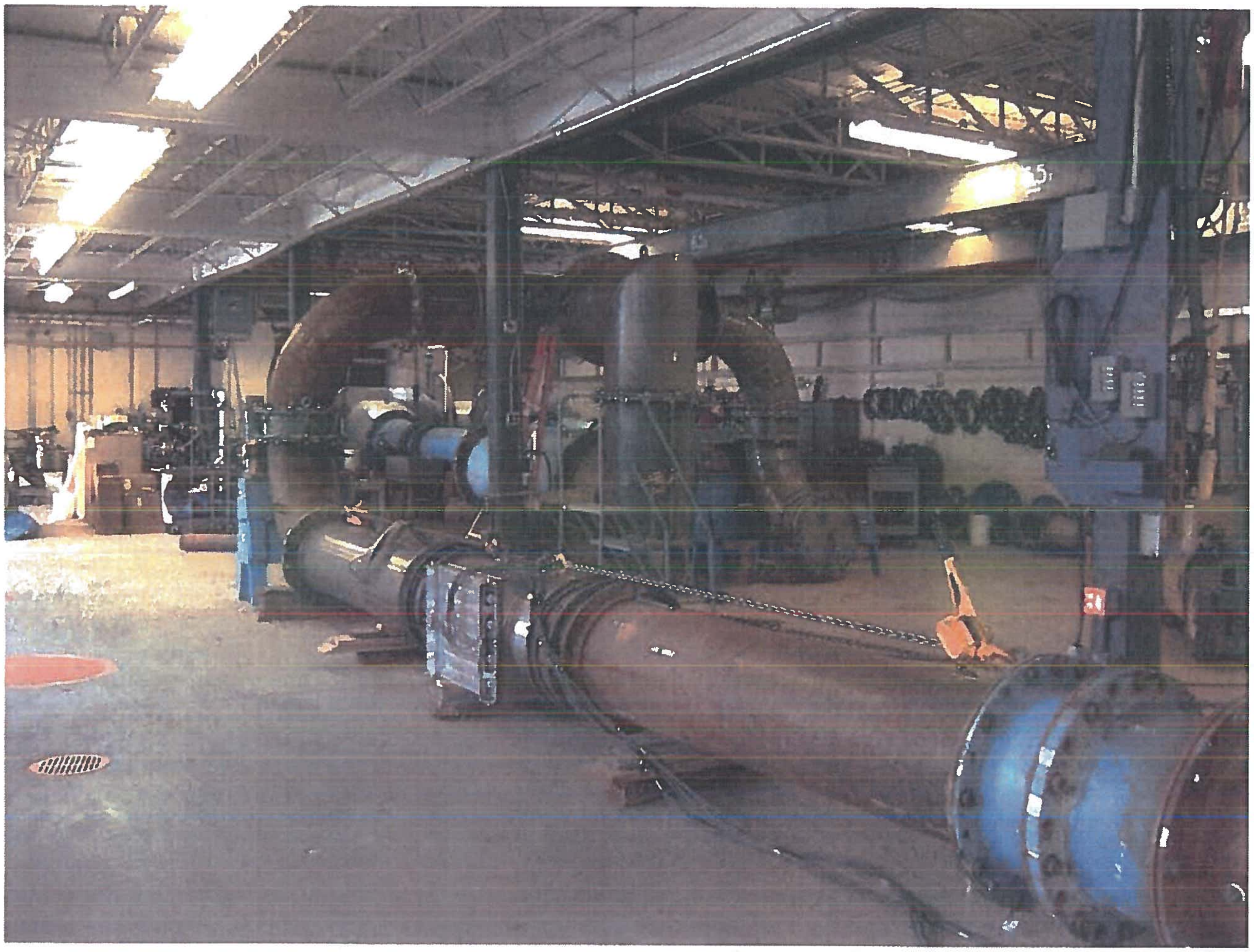

Figure 3: Unit 1 Loop A - Hydraulic Model

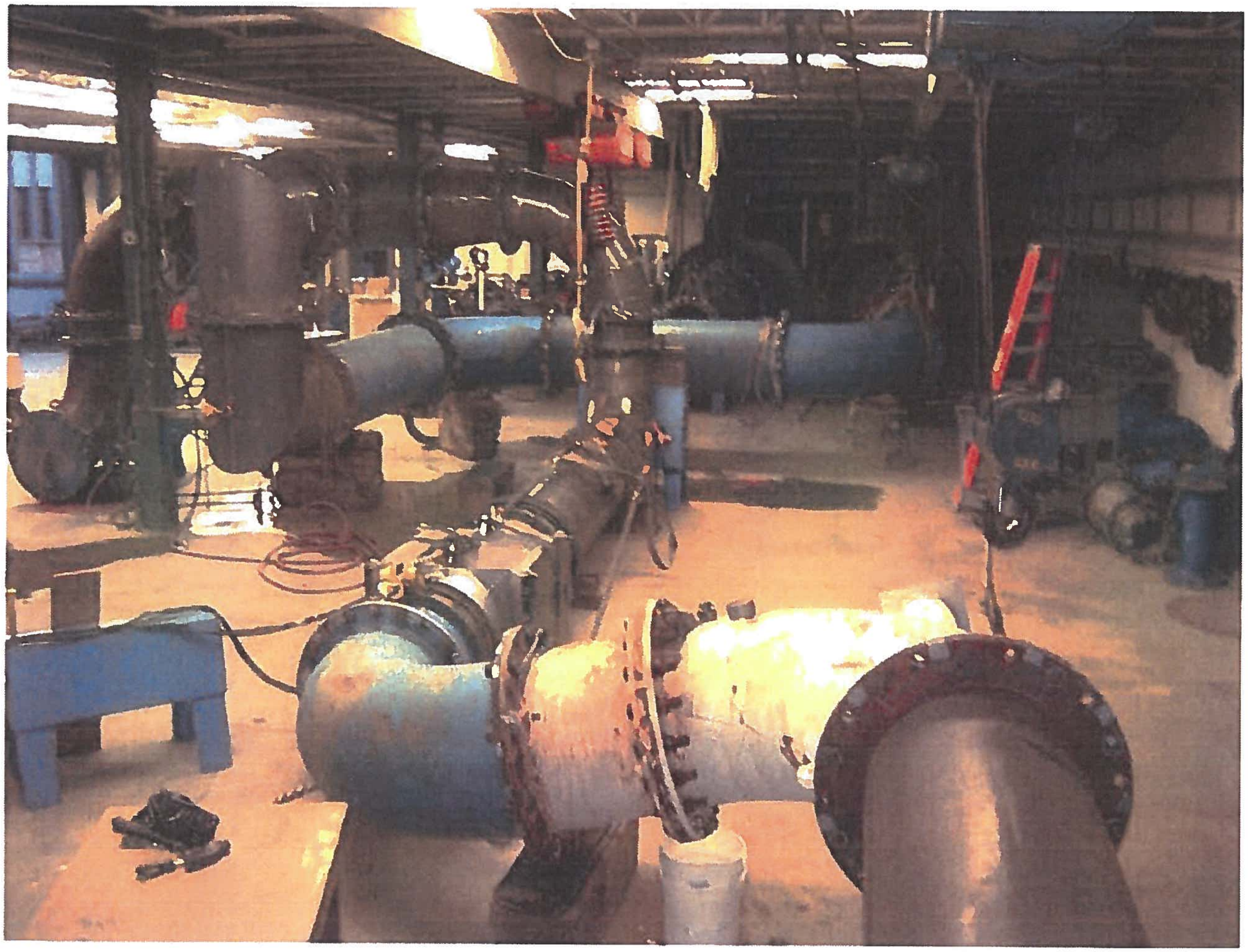

Figure 4: Unit 3 Loop A - Hydraulic Model

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Figure 5: [ ]

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2.3 Calibration Data and Model Tests

Reference 1 outlines the model tests performed. These tests included variations to the model, such as changes to the upstream hydraulics. With these tests, installation uncertainty is addressed. [

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The Oconee parametric tests and calibration numbers for Units 1, 2 and 3 are described in Tables 3, 4 and 5, respectively. The Oconee test data is shown Appendix A. [

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Table 3: Unit 1 Test Summary

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Table 4: Unit 2 Test Summary

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Table 5: Unit 3 Test Summary

2.3.1 Test Collection Procedure

Weigh tank testing at a specific flow rate began by setting the proper flow in the flow loop using a remotely operated butterfly valve located downstream of the model. The flow meter data is collected using a serial connection to a laptop PC. The flow data is polled by the PC from the electronics at approximately 5-second intervals. The ported information contains a time stamp and volumetric flows, as well as signal data quality, and path velocity data. Velocity data for individual acoustic paths are recorded in order to evaluate the fluid velocity profile.

The test procedure at any given flow rate was as follows:

- Set the flow rate and allow flow and temperature to stabilize
- Alden personnel operate weigh tank run by moving the diverter valve.
- Cameron personnel separately record a start time and a stop time along. This information is used to synchronize the ported data with the weigh tank data.

3.0 METER FACTOR CALCULATION

3.1 Meter Factor Definition

The meter factor accounts for (typically small) biases in the numerical integration due to the hydraulics, dimension measurements and acoustics of the application. The flow meter software multiplies the result of the multi-path numerical integration by the product of the meter factor to obtain the flow rate. For the Alden tests, the meter factor was set at 1.000.

The meter factor is calculated by the following equation:

$$MF = \frac{Q_{Alden}}{Q_{meter}}$$

Where:

Q_{meter} = Volumetric flow rate from meter (with meter factor set to 1.000)

Q_{Alden} = Volumetric flow rate based on Alden weigh tank

3.2 Test Results

Appendix A tabulates the results of the individual calibration runs. Each tab/subsection of Appendix A documents the different hydraulic configurations. All tables contain for each weigh tank run:

- Alden certified flow rate.
- Meter flow rate averaged over the weigh tank fill period [ ]
- [ ]

Tables 6, 7 and 8 below summarize the data [ ] for Units 1, 2 and 3, respectively.

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Figures 6 - 11 plot the meter factor for all the tests (note: for clarity only the average meter factor at each flow rate is plotted).

Table 6: [ ][4]
for Oconee Unit 1

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Table 7: [ ][4]
for Oconee Unit 2

[4] [ ]

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Table 8: [ ][4]
for Oconee Unit 3

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Figure 6: [ ]

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Figure 7: [ ]

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Figure 8: [ ]

Figure 9: [ ]

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Figure 10: [ ]

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Figure 11: [ ]

3.3 Measured Velocity Profiles

Using the flow meter it is possible to measure and understand the actual hydraulic velocity profile. Appendix A has the velocity profiles observed during each model test. [

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Figure 12: [ ]

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Figure 13: [ ]

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Figure 14: [ ]

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Figure 15: [ ]

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Figure 16: [ ]

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Figure 17: [ ]

3.3.1 *Swirl Rate Definition*

Swirl can be measured by the flow meter. Cameron quantifies swirl rate with a swirl rate calculation, as follows:

$$\text{Swirl Rate} = \text{Average}\left[\frac{V_1 - V_5}{2 \cdot y_S}, \frac{V_8 - V_4}{2 \cdot y_S}, \frac{V_2 - V_6}{2 \cdot y_L}, \frac{V_7 - V_3}{2 \cdot y_L}\right]$$

Where:

V_1, V_4, V_5, V_8	=	Normalized velocities measured along outside chords
V_2, V_3, V_6, V_7	=	Normalized velocities measured along inside chords
y_S , y_L	=	Normalized chord location for short and long paths

Swirl rates less than 3% are low and are typically observed in models with only planar connections. Swirl rates greater than 3% are considered "swirling". Swirl rates greater than 10% are considered to have strong swirl.

3.3.1.1 *Swirl Rate Results*

The following figure summarizes the absolute value of swirl rate observed during the calibrations.
[ ]

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Figure 18: [ ]

3.3.2 Flatness Ratio Definition

Cameron uses the flatness ratio (FR) to quantify the flatness of the velocity profile. FR is defined as:

$$FR = \left[\frac{V_1 + V_4 + V_5 + V_8}{V_2 + V_3 + V_6 + V_7}\right]$$

Where:

V_1, V_4, V_5, V_8	=	velocities measured along the outside chords (or short paths)
V_2, V_3, V_6, V_7	=	velocities measured along the inside chords (or long paths)

When a velocity profile is perfectly flat, then FR equals 1.0. When a velocity profile is laminar, then the FR equals approximately 0.38. The limits of 0.38 and 1.0 represent extremes. The FR is a function of Reynolds number but also is strongly influenced by the hydraulics upstream of the flow meter.

Typical feedwater applications have FR in the range of 0.78 to 0.95. Downstream of flow conditioners, the velocity profile tends to be pointier and the FR value is lower, 0.78 to 0.80. Downstream of elbows and tees the velocity profile tends to be flatter and the FR value is higher, 0.85 to 0.95. The actual range at a given plant is dependent upon site upstream conditions (for example the hydraulic fittings such as tees, elbows, etc.). The tests are summarized in the following figure.

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Figure 19: [ ]

3.4 Relationship between Flatness Ratio and Meter Factor

In 2002, Cameron published an analysis of velocity profiles observed in the field. In this analysis, an analytical relationship between the meter factor (MF) and the observed flatness ratio (FR) was computed. [

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Figure 20: [ ]

[

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Figure 21: [ ]

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4.0 METER FACTOR ACCURACY ASSESSMENT

This section documents the methodology for calculating the uncertainty or accuracy of the meter factor. This report was produced using a process and quality assurance consistent with the requirements of ASME PTC 19.1 and ANSI/NCSL Z540-2-1997 (see References 6, 10, 11, 16 and 17). The approach to determination of the set points is to combine the random (Type A) and systematic (Type B) terms by the means of the RSS approach given that all the terms are independent, zero-centered and normally distributed.

First, the sensitivity of the calculated flow to each independent variable or input is determined. Once the sensitivities to the independent variables have been calculated, then the independent variables' uncertainties are calculated and multiplied with their sensitivity coefficient, such as calibration facility, timing errors, etc. The 95% confidence level uncertainty bounds are calculated for each element (uncertainty coverage for each term is 95%).

The evaluation of the sensitivity coefficients is performed by determining the independent variables in the mass flow (and volumetric flow) calculation. For example, if volume flow is a function of independent variables $X_1, X_2, \ldots, X_n$, as follows:

$$Q = f(X1, X2, \ldots, Xn)$$

The uncertainty effect of specific independent variables on the flow measurement is calculated by partial differentiation of the above equation. Expressing the result as a per unit sensitivity:

$$\frac{dQ}{Q} = \left[\frac{X_1 \partial Q}{Q \partial X_1}\right]\left(\frac{\Delta X_1}{X_1}\right) + \left[\frac{X_2 \partial Q}{Q \partial X_2}\right]\left(\frac{\Delta X_2}{X_2}\right) + \ldots\ldots + \left[\frac{X_n \partial Q}{Q \partial X_n}\right]\left(\frac{\Delta X_n}{X_n}\right).$$

Where the terms in the brackets are the sensitivity coefficients for $X_1, X_2, \ldots, X_n$. The magnitudes and signs of each uncertainty for a given flow measurement are then bounded by 95% confidence intervals.

ASME PTC 19.1 demonstrates that by combining the independent uncertainty contributions as the root sum square, the overall uncertainty in volumetric flow is bounded by a 95% confidence level.

The allocation of uncertainties for meter factor for the flow meter (consistent with the Cameron Topical report) is shown in Tables 9 - 11 below. Using the data in these tables and the root mean square summation technique indicated for combining independent uncertainties of relatively the same magnitude, the total uncertainty due to MF is computed.

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Table 9: Uncertainty Summary for Unit 1 Meter Factor

RMS values rounded to closest 0.001%

(All terms are treated as normal distributions with k = 2, e.g., 95% coverage)

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Table 10: Uncertainty Summary for Unit 2 Meter Factor

RMS values rounded to closest 0.001%

(All terms are treated as normal distributions with k = 2, e.g., 95% coverage)

[5] For random (Type A) terms, the system uncertainty is the root sum square of the individual uncertainties weighted by their flow relative to the total flow (e.g., one third). For systematic terms (Type B), the system uncertainty is weighted average of terms.

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Table 11: Uncertainty Summary for Unit 3 Meter Factor

RMS values rounded to closest 0.001%

(All terms are treated as normal distributions with k = 2, e.g., 95% coverage)

4.1 Facility Uncertainty

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A facility uncertainty of [ ][6] has been budgeted and this figure appears in the table above.

4.2 Measurement Uncertainty

Appendix B calculates the uncertainties in the volumetric flow measurement (excluding meter factor) of the flow meters used for this test. The results are summarized below in Table 6. [ ]

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For this report, a typical flow of 8,500 gpm (1931 CMH) was used (this is the average rate of Loop A, Test A-1 and it is comparable to the other model tests).

[6] See Reference 14.

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Table 12: Uncertainties in Volumetric Flow Measurements
(All Figures rounded to three decimal points)
(All terms are treated as normal distributions with k = 2, e.g., 95% coverage)

4.3 Extrapolation - Profile Variation Allowance

At the plant, it is likely that the hydraulic conditions will not equal those tested during the calibration. In particular, the plant's Reynolds numbers are higher than that achievable at the laboratory (approximately 20 million vs. ~2 million). Further, the plant may have a lower wall roughness than the test pipes used at the laboratory.

[ ]

[ ]

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The numerical calculation of meter factor is illustrated below.

Figure 22: [ ]

Figure 23: [ ]

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Using this analysis, a meter factor extrapolation of [ ] is predicted from the average calibration Reynolds number (~2e6) to the plant application (~20e6). The flatness ratio extrapolation is approximately [ ] for the same range. Trade Secret & Confidential Commercial Information

[ ] Trade Secret & Confidential Commercial Information

4.4 Modeling Sensitivity Uncertainty

For modeling sensitivity, Cameron used the similar hydraulic models for all three Oconee Units. Parametric tests were constructed so that a large range of hydraulic conditions could be tested across all the Oconee models. Using this larger data population, a robust statistical analysis is possible.

The approach (identical to that discussed in Reference 19) is [ ] the modeling sensitivity is presented for Oconee in the following table: Trade Secret & Confidential Commercial Information

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4.5 Mean Meter Factor Uncertainty

Each meter factor is computed as the average (mean) of the meter factor measurements made for that flow element. The uncertainty in mean meter factor addresses the 95% confidence limits on the uncertainty in that mean. Each set of data at a given flow rate is treated as a separate datum, since in fact the profile varies with flow rate (i.e., Reynolds Number) as shown in Section 4.3. Section 4.4 shows that meter factor is essentially independent of hydraulic configuration. Hence the precision with which the meter factor for a specific flow element is determined is enhanced by including all calibration data for that flow element. Accordingly, the meter factor is determined from [ ] Trade Secret & Confidential Commercial Information

[7] [ ]

[8] [ ] Trade Secret & Confidential Commercial Information

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The calculation of the uncertainty of the mean proceeds as follows:

a. [

b.

c.

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Using the above methodology, the uncertainty in the mean meter factor is computed to be as follows:

- [ ]
- [ ]
- [ ]
- [ ]
- [ ]
- [ ]

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5.0 REFERENCES

1. ALD-1134 Rev 0, Hydraulic Calibration Plan for Oconee Units 1, 2 and 3, 24" Chordal Spool Pieces
2. 2006 South East Asia Flow Workshop Paper, "The Relative Merits of Ultrasonic Meters Employing Between Two and Eight Paths", Gregor Brown, Don Augenstein, Terry Cousins, Herb Estrada
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5. Moody, L. F., "Friction Factors for Pipe Flow," ASME Transactions, V. 66, 1944, pp. 671-694
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7. Murakami, M., Shimizu, Y., and Shiragami, H., "Studies on Fluid Flow in Three-Dimensional Bend Conduits," Japan Society of Mechanical Engineering (JSME), Bulletin V. 12, No. 54, Dec. 1969, pp. 1369-1379.
8. Cameron Topical Report ER-80P Rev 1, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System", March 1997.
9. ER-551, Transducer Replacement Sensitivity
10. ASME PTC 19.1-1985, Measurement Uncertainty
11. Cameron Engineering Report ER-160P Rev 1, "Supplement to Topical Report ER 80P: Basis for a Power Uprate with the LEFM System", May 2000
12. Cameron Engineering Report ER-157P Rev. 5, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus", October 2001
13. Alden Calibration Report
14. Cameron Engineering Report ER 262 Rev 1, "Effects of Velocity Profile Changes Measured In-Plant on LEFM Feedwater Flow Measurement Systems", January 2002
15. NIST TN-1297, "Guidelines for Evaluating and Expressing the Uncertainty of NIST Measurement Results"
16. ANSI/NCSL Z540-2-1997, "U.S. Guide to the Expression of Uncertainty in Measurement"
17. ISO "Guide to the Expression of Uncertainty in Measurement"
18. IGHEM Flow Paper 2008 Milan, Italy, "Accuracy Validation of Multiple Path Transit Time Flowmeters"
19. Westinghouse Research Laboratories Nov 3, 1972, "Integration Errors for Turbulent Profiles in Pipe Flow"

Appendix A – Calibration Data

This Appendix contains the raw data for each test. The data includes the Alden calibration period flow, the average flow during the calibration, and the computed meter factor at each flow.

No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix B - Meter Uncertainty

B.1 – Inputs and Scaling
B.2 – Flow Uncertainty
B.3 – [ ]
B.4 – [ ]

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Appendix B.1

Inputs and Scaling

No attachment to follow, as Appendix is Proprietary in its Entirety

Appendix B.2

Flow Uncertainty

No attachment to follow, as Appendix is Proprietary in its Entirety

Appendix B.3

[ ]

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No attachment to follow, as Appendix is Proprietary in its Entirety

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Appendix B.4

[ ]

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No attachment to follow, as Appendix is Proprietary in its Entirety

ATTACHMENT 7 CAMERON AFFIDAVITS

Application for Withholding Proprietary Information from Public Disclosure and covering ER-813P Rev 6, ER-824P Rev 6, and ER-825P Rev 6

Application for Withholding Proprietary Information from Public Disclosure and covering ER-855 P Rev 0

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Cameron Technologies US LLC

January 2, 2020
CAW 20-01

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject:

1. Caldon® Ultrasonics Engineering Report: ER-813P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Nuclear Generating Station Using the LEFM✓+ System," November 2019
2. Caldon® Ultrasonics Engineering Report: ER-824P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Nuclear Generating Station Using the LEFM✓+ System," November 2019
3. Caldon® Ultrasonics Engineering Report: ER-825P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Nuclear Generating Station Using the LEFM✓+ System," November 2019

Gentlemen:

This application for withholding is submitted by Cameron Technologies US, LLC, a Delaware limited liability company (herein called "Cameron") 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 20-01 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 20-01 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

January 2, 2020
CAW 20-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Joanna Phillips, who, being by me duly sworn according to law, deposes and says that she is authorized to execute this Affidavit on behalf of Cameron Technologies US, LLC, a Delaware limited liability company (herein called "Cameron"), and that the averments of fact set forth in this Affidavit are true and correct to the best of her knowledge, information, and belief:

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 2nd day of

January, 2020

Frances A Lewis
Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

1. I am the Nuclear Sales Manager for Caldon Technologies US, LLC, 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:

1. Caldon® Ultrasonics Engineering Report: ER-813P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Nuclear Generating Station Using the LEFM✓+ System," November 2019
2. Caldon® Ultrasonics Engineering Report: ER-824P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Nuclear Generating Station Using the LEFM✓+ System," November 2019
3. Caldon® Ultrasonics Engineering Report: ER-825P Rev 6 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Nuclear Generating Station Using the LEFM✓+ System," November 2019

- Pages 3, 4, 5, 7, 8, 10 contain partial proprietary information
- Appendix A.4 and A.5 cover pages contain partial proprietary information
- Appendices A.1, A.2, A.4, A.5, and B are proprietary in their entirety

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Oconee Unit 1 for flow measurement at the licensed reactor thermal power level of 2610 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.

Cameron Technologies US LLC

January 2, 2020
CAW 20-02

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon® Ultrasonics Engineering Report: ER-855P Rev 0 "Meter Factor Calculation and Accuracy Assessment for the LEFM Check Plus Meters at Oconee Units 1, 2, and 3," November 2019

Gentlemen:

This application for withholding is submitted by Cameron Technologies US, LLC, a Delaware limited liability company (herein called "Cameron") 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 20-02 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 20-02 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

January 2, 2020
CAW 20-02

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Joanna Phillips, who, being by me duly sworn according to law, deposes and says that she is authorized to execute this Affidavit on behalf of Cameron Technologies US, LLC, a Delaware limited liability company (herein called "Cameron"), and that the averments of fact set forth in this Affidavit are true and correct to the best of her knowledge, information, and belief:

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 2nd day of

January, 2020

Frances A. Lewis

Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

1. I am the Nuclear Sales Manager for Caldon Technologies US, LLC, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

5. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Cameron.

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a

system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Caldon® Ultrasonic Engineering Report: ER-855P Rev 0 - "Meter Factor Calculation and Accuracy Assessment for the LEFM Check Plus Meters at Oconee Units 1, 2, and 3," November 2019

- Pages 1, 2, 3, 4, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 30, 31, 32, 34, 35, 36, 37, 38, 39 contain partial proprietary information
- Appendix B index page and Appendix B.3 and B.4 cover pages contain partial proprietary information
- Appendices A, B.1, B.2, B.3, and B.4 are proprietary in their entirety

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 Systems used by Oconee Units 1, 2, and 3 for flow measurement at the licensed reactor thermal power level of 2610 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.