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10 CFR 50.90

September 20, 2011

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2 and 3
Renewed Facility Operating Licenses Numbers DPR-38, -47, -55;
Docket Number 50-269, 50-270 and 50-287;
License Amendment Request for Measurement Uncertainty Recapture Power Uprate,
License Amendment Request No. 2011-02

Reference: Regulatory Issue Summary 2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002

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, 47 and 55 to support a measurement uncertainty recapture (MUR) power uprate. This MUR 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. (The allowable increase of 1.66% Rated Thermal Power truncated to the lower whole megawatt is a 1.64% uprate.) 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 required by 10 CFR 50, Appendix K by 1.64%. Specifically, this LAR requests NRC approval for certain Oconee Nuclear Station (ONS) Technical specification changes necessary to support operation at the uprated power level.

The following enclosures and attachments are provided to support the proposed TS changes:

Enclosure 1	Evaluation of the proposed changes
Enclosure 2	Technical review of the proposed uprate, in the format of RIS 2002-03
Attachment 1	List of Regulatory Commitments
Attachment 2	Marked up Technical Specification pages
Attachment 3	Marked-up Technical Specification Bases pages (for information only)

Attachment 6 to this letter contains proprietary information.
Withhold From Public Disclosure Under 10 CFR 2.390.
Upon removal of Attachment 6, this letter is uncontrolled.

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Attachment 4	Reprinted Technical Specification pages
Attachment 5	Reprinted Technical Specification Bases pages (for information only)
Attachment 6	Heat Balance Uncertainty Analyses (Oconee specific)

Attachment 6 includes Cameron documents containing information that has been classified as proprietary by Cameron. An affidavit from Cameron for those documents considered proprietary is also included in Attachment 6. This affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC pursuant to 10 CFR 2.390. Attachment 6 also includes a Duke Energy document containing information that has been classified as proprietary by Duke Energy. An affidavit from Duke Energy for the document considered proprietary is also included in Attachment 6. This affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC pursuant to 10 CFR 2.390.

Regulatory evaluation (including the significant hazards consideration) and environmental considerations are provided in Sections 5 and 6 of Enclosure 1. Attachment 1 provides a list of regulatory commitments being made as a result of this LAR.

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

Duke Energy requests approval of this amendment request by March 15, 2012. Once approved, the amendment will be implemented within 120 days. Duke Energy will also update applicable sections of the ONS Updated Final Safety Analysis Report (UFSAR), as necessary, and submit these per 10 CFR 50.71(e).

Inquiries on this proposed amendment request should be directed to Boyd Shingleton of the Oconee Regulatory Compliance Group at (864) 873-4716.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 20, 2011.

Sincerely,

TPGillespie

T. Preston Gillespie, Jr., Vice President,
Oconee Nuclear Station

Enclosures and Attachments:

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cc w/attachments:

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ENCLOSURE 1

EVALUATION OF PROPOSED CHANGES

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

1. SUMMARY DESCRIPTION
2. BACKGROUND
3. DETAILED DESCRIPTION OF PROPOSED CHANGES
4. TECHNICAL EVALUATION
5. REGULATORY EVALUATION
   - Significant Hazards Consideration
   - Applicable Regulatory Requirements/Criteria
   - Precedent
   - Conclusion
6. ENVIRONMENTAL CONSIDERATION
7. REFERENCES

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 Nos. DPR-38, -47 and -55 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.

Selective Licensee Commitments (SLCs) and the UFSAR will be changed as required to support the power uprate in accordance with 10 CFR 50.59 following implementation of the MUR uprate.

2 BACKGROUND

Oconee Units 1, 2 & 3 are presently licensed for a core power rating of 2568 MWt. Through the use of more accurate feedwater flow measurement instrumentation, Duke Energy is seeking to increase the licensed core power to 2610 MWt, an increase of 1.64%.

The 1.64% core power uprate for Oconee Units 1, 2 & 3 (hereby referred to as the MUR Power Uprate) is based on recapturing measurement uncertainty currently included in the analytical margin originally required for 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. 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 reduce instrument uncertainty; core power level can be determined with a power measurement uncertainty of approximately 0.34 percent. Thus, Duke Energy proposes to reduce the licensed power uncertainty by 1.64%.

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.

3 DETAILED DESCRIPTION OF PROPOSED CHANGES

To accommodate a rated thermal power level of 2610 megawatts thermal for Oconee Units 1, 2, and 3, Duke Energy proposes to modify the Operating License, Technical Specifications and Technical Specification Bases (for information only). The proposed changes are listed below:

TS 1.1 Definition of Rated Thermal Power

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

TS Table 3.3.1-1, Reactor Protective System Instrumentation

The Nuclear Overpower high setpoint allowable value (line item 1a) is currently 105.5% of licensed RTP (105.5% of 2568 MWt = 2709 MWt), and will remain at 105.5% of the uprated licensed RTP (105.5% of 2610 MWt = 2754 MWt). This allowable value (line item 1a) will be applicable when all four reactor coolant pumps are in operation. The trip setpoint was left at 105.5% RTP (2754 MWt) as that value is bounded by the value assumed in the accident analyses for Oconee Nuclear Station.

Line item 1b will be added and will be the Nuclear Overpower high setpoint allowable value (79.3% RTP) when only three reactor coolant pumps (RCP) are operating. The basis for the new setpoint is to better mitigate the UFSAR Chapter 15.17 Small Steam Line Break transient with 3 RCPs in operation.

The previous low setpoint (line item 1b) is unchanged except that it is now line item 1c.

Although not directly related to the MUR, footnotes d and e are added to each entry of Surveillance Requirement 3.3.1.2 in the table and notes d and e are added at the bottom of the table. These notes make the table consistent with Technical Specification Task Force document TSTF 493.

Note f was added to state "If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in the Selected Licensee Commitments."

TS 3.4.4 RCS Loops – Mode 1 and 2

LCO 3.4.4.b will change to say that thermal power will be restricted to 73.8% RTP for two reactor coolant system (RCS) loops in operation with three reactor coolant pumps (RCPs) operating, and to add corresponding LCO actions. The nominal 3 RCP operation power level assumed in the safety analyses is unchanged for the MUR uprate. The current nominal power level is 75% of 2568 MWt, which will become 73.8% of 2610 MWt, or 1926 MWt.

Operating Licenses Page 3 – Maximum Power Level

For each of the three 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.

Selected Licensee Commitments

A Selected Licensee Commitment (SLC) is being added to support 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. The SLC changes are not provided as part of this LAR, but are being controlled using the 10 CFR 50.59 process. An SLC is also provided for implementation of Technical Specification Task Force Improved Standard Technical Specification Change Traveler TSTF-493.

4 TECHNICAL EVALUATION

Oconee Units 1, 2 and 3 are presently licensed for an 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 addressed the following categories: the feedwater flow measurement technique and power measurement uncertainty, accidents and transients that remain bounded at the higher power level, accidents and transients that are not bounded at the higher 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 any changes to the Technical Specifications including protective system setpoints. The evaluation conclusions are summarized in Enclosure 2, in the format of NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 1).

In addition, Duke Energy evaluated the potential impact of pending/future LARs on the MUR evaluation (or vice versa). The following LAR topics were included in this evaluation:

1) Change to Reactor Vessel Inspection Plan,
2) Tornado/High Energy Line Break,
3) Main Steam Isolation Valves and
4) Protected Service Water (NFPA-805).

The MUR has no impact on Items 1, 3, or 4 above. The impact of the High Energy Line Break (HELB) LAR (item 2) is addressed in Enclosure 2, Section III. Item 3 is the only future LAR. Duke Energy committed to add Main Steam Isolation Valves (MSIVs - #6 above) in the HELB LAR and associated supplements. The proposed MSIVs are necessary to achieve safe shutdown following certain MSLBs when using the SSF for event mitigation. Any impact of adding MSIVs will be addressed by the MSIV LAR.

5 REGULATORY EVALUATION

5.1 Significant Hazards Consideration

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment to ONS 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. It will also reduce some protective setpoints when expressed as a percent of rated thermal power. All Technical Specification 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 but two analyses remain bounding at the uprated power of 2610 MWt. These two analyses (High Energy Line Break and Double Main Steam Line Break) were reanalyzed at the higher power level and found to be acceptable.

   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 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.

   As summarized in Sections IV, V and VI of Enclosure 2, the proposed change does not involve any change to the design or functional requirements of the associated 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 change demonstrates that the availability 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 changes do 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

   A Failure Modes and Effects Analysis of the new system was performed, and the possible effects of failures of the new equipment and the increased power level on the overall plant systems were reviewed. This review found that no new or different accidents were created by the new equipment or the uprated power levels.

No installed equipment is being operated in a different manner. The proposed changes have no significant adverse affect on any safety-related SSCs and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. The uprated power does not create any new accident initiators. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed changes.

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 operating power level of the plants, it retains the 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 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 Oconee safety analyses, and the revised design basis radiological accident dose calculations, bound the power uprate and therefore do not significantly impact margins.

   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.

5.2 Applicable Regulatory Requirements/Criteria

Regulatory Issue Summary (RIS) 2002-03 provides generic guidance for evaluating a MUR power uprate. Enclosure 2 to this license amendment request 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 be in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5.3 Precedent

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

1 FirstEnergy Nuclear Operating Company submittal for MUR power uprate of Davis-Besse Nuclear Power Station, which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated Jun4 30, 2008 (TAC No. MD8326)

2 Progress Energy submittal for MUR power uprate of Crystal River Nuclear Plant, Unit 3, which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated December 26, 2007 (TAC No. MD5500)

3 Virginia Electric and Power Company submittal for MUR power uprate of the Surry Power Station, Units 1 and 2, which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated September 24, 2010 (TAC Nos. ME3293 and ME3294)

4 AEP, Indiana Michigan Power submittal for MUR power uprate of the Donald C. Cook Nuclear Plant Unit 1, which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment dated December 20, 2002 (TAC No. MB5498)

5.4 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 5.2 above.

Duke Energy identified several LARs, as indicated in Section 5.3 above, requesting MUR power uprates. These LARs used the applicable regulatory requirements of Section 5.2 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.

6 ENVIRONMENTAL CONSIDERATION

Duke Energy has evaluated this LAR 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 LAR 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 5.1 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).

7 REFERENCES

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

ENCLOSURE 2

SUMMARY OF RIS 2002-03 REQUESTED INFORMATION

This enclosure provides responses to RIS 2002-03, Attachment 1, with the Oconee Nuclear Station (ONS) information provided in response to each item.

TABLE OF CONTENTS for Enclosure 2:

		Page
Title Page		E2-i
ACRONYMS		E2-ii
Section I	Feedwater flow measurement technique and power level measurement uncertainty	E2-1
Section II	Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level	E2-11
Section III	Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level	E2-39
Section IV	Mechanical structural material component integrity and design	E2-43
Section V	Electrical equipment design	E2-64
Section VI	System design	E2-69
Section VII	Other	E2-81
Section VIII	Changes to Technical Specifications, protection system settings, and emergency system settings	E2-86

ACRONYMs:

AC	Alternating current
ADV	Atmospheric Dump Valve
AFIS	Automatic Feedwater Isolation System
ALARA	As Low As Reasonably Achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
AOR	Analysis of record
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
CCW	Condenser Circulating Water
CF	Core Flood
CFM	Centerline Fuel Melt
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRVS	Control Room Ventilation System
CS	Coolant Storage
CUF	Cumulative usage factor
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
EDB	Equipment database

EFPY	Effective full-power years
EFW	Emergency Feedwater
EFWPT	Emergency feedwater pump turbine
EMA	Equivalent margins analysis
EMS	Emergency Makeup System
EQ	Environmental Qualification
ES	Engineered Safeguards
FAC	Flow-accelerated corrosion
FDW	Main feedwater
FIV	Flow induced vibration
FWLB	Feedwater line break
FWPT	Feedwater pump turbine
GWD	Gaseous Waste Disposal
HELB	High energy line break
HHASW	High Head Auxiliary Service Water System
HPI	High Pressure Injection
HPSW	High Pressure Service Water
HZP	Hot Zero Power
ICS	Integrated Control System
IPB	Isolated Phase Bus
ISA	Instrument Society of America
ISLH	Inservice leak and hydrostatic
IST	Inservice Testing
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)
LOCA	Loss of Coolant Accident
LOMFW	Loss of main feedwater
LOOP	Loss of Offsite Power
LPI	Low Pressure Injection
LPSW	Low Pressure Service Water
LRSS	Locked Rotor, Shaft Seizure accident
LTOP	Low temperature overpressure protection
LWD	Liquid Waste Disposal
M&E	Mass and energy
MFW	Main Feedwater
MFWLB	Main Feedwater Line Break
MS	Main Steam
MSIV	Main steam isolation valve

MSLB	Main Steam Line Break
MTC	Moderator Temperature Coefficient
MUR	Measurement Uncertainty Recapture
MVAR	1,000,000 VARs
MW	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
OBDN	Operable But Degraded/Non-Conforming
ODCM	Offsite Dose Calculation Manual
ONS	Oconee Nuclear Station (Units 1, 2, and 3)
OP	Operating Procedures
OTSG	Once Through Steam Generator
PAS	Post Accident Sampling
PIP	Problem Investigation Process
PRP	Reactor Building Purge
PSW	Protected Service Water
PTC	Performance Test Code (an ASME document)
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
SER	Safety Evaluation Report
SFP	Spent fuel pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture

SLC	Selected Licensee Commitments
SRSS	Square Root of the Sum of the Squares
SSCs	Systems, Structures and Components
SSF	Standby Shutdown Facility
TBS	Turbine Bypass System
TDEFWP	Turbine driven emergency feedwater pump
TID	Total integrated dose
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USE	Upper shelf energy
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 technique used at Oconee Units 1, 2, and 3 is a Cameron (aka Caldon) CheckPlus Leading Edge Flow Meter (LEFM) with ultrasonic multi-path transit time flow meter as described in the following topical reports.

Cameron Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," Revision 0, March, 1997

Cameron Engineering Report ER-157(P-A), "Supplement to Cameron Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or CheckPlus," Revision 8, May 2008

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) were reviewed and approved by the NRC in the SER in letter 2 below.

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

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.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 consists of an electronic cabinet and two measurement section/spool pieces (consisting of four electronic transmitters and four pressure transmitters). One measurement section/spool piece will be installed in each of the two 24 inch main feedwater flow headers that feed each steam generator. The measurement sections are located upstream of the existing feedwater flow venturis.

The location for the Oconee Unit 1 LEFMs meets all the Cameron requirements for LEFM location. The Oconee Unit 1 LEFMs are installed in horizontal runs of main feedwater piping upstream of the existing venturis. The LEFMs meet or exceed the required 5 L/D (length / diameter) downstream of elbows, laterals, or headers.

The exact locations of the Oconee Units 2 and 3 LEFMs will be determined as the Engineering Change packages for those installations are developed. Duke Energy commits (See Attachment 1 to this LAR) that the Unit 2 and Unit 3 LEFMs will also be positioned such that they meet all Cameron requirements and thus will have no effect on the existing venturis.

The location of the LEFMs relative to the venturi was reviewed and it was determined that the LEFM locations will not affect the existing venturi performance. Cameron recommends the LEFMs be located at least 4 L/D above the existing venturi to ensure no effect, and the Oconee Unit 1 A and B LEFMs are 11.2 and 13.5 L/D upstream of the venturis, respectively.

Testing of each of the Oconee LEFM CheckPlus system was performed at Alden Research Laboratories and the results are documented in Cameron Engineering Report ER-855 (Reference I.9), which is included in Attachment 6 to this LAR. Separate piping arrangements (shown in figures 1 and 2 of ER-855) were used for Oconee Unit 1 and Oconee Units 2 and 3. 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 Topical Report ER-80P, the NRC established four criteria to be addressed by each licensee. In approving Caldon Topical Report ER-157P, Revision 8, the NRC established five additional criteria to be addressed by each licensee. A discussion of each of the nine criteria relative to Oconee Units 1, 2, & 3 follow:

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.

Response to Criterion 1:

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 preventative 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 preventative 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 CheckPlus System ensures that the system remains bounded by the analysis and assumptions set forth in the Topical Report ER-80P. The incorporation of, and continued adherence to, these requirements will assure that the LEFM CheckPlus system is properly maintained and calibrated. Duke Energy's commitment to complete this maintenance program is included in Attachment 1 to this LAR.

Operation:

Details of Oconee's proposed operation with the LEFM not fully functional are discussed in response to Criterion 1 from ER-157P, Revision 8, below.

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

Response to Criterion 2:

Criterion 2 does not apply to Oconee Units 1, 2, & 3 as they do not have LEFMs installed at this time.

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.

Response to Criterion 3:

The LEFM uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1, Instrument Society of America (ISA) Recommended Practice (RP) ISA RP 67.04 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.

The feedwater flow and temperature uncertainties are combined with other plant measurement uncertainties (steam temperature, steam pressure, feedwater pressure) to calculate the overall heat balance uncertainty. The heat balance uncertainty calculation using the LEFMs is consistent with the current heat balance uncertainty calculation that uses the feedwater flow venturis and RTDs. The current calculation is based on a square-root-of-the-sum-of-the-squares (SRSS) calculation.

Criterion 4 from ER-80P - For plants 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 installation), additional justification should be provided for its use. The justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4:

This criterion does not apply to Oconee, as the flow elements were tested and calibrated in a full-scale model of the Oconee Units 1, 2, & 3 hydraulic geometry at the Alden Research Laboratory. A bounding calibration factor for the Oconee Units 1, 2, & 3 spool pieces was established by these tests and is included in the Cameron engineering reports for each unit. (Cameron reports ER-813 for ONS-1, ER-824 for ONS-2, and ER-825 for ONS-3 are included in Attachment 6 to this LAR). A Cameron engineering report (ER-855 is included in Attachment 6 to this LAR) summarizes the testing and evaluates the test data. A bounding uncertainty for the LEFM has been provided for use in the uncertainty calculation described in Section I.1.E below. A copy of the site-specific uncertainty analysis is in Attachment 6 to this License Amendment Request.

Final acceptance of the ONS specific uncertainty analysis will occur after completion of the commissioning process, which verifies that in-situ test data is bounded by the calibration test data.

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.

Response to Criterion 1:

An engineering evaluation was performed to justify an allowed outage time upon loss of the LEFM signal. This evaluation is based on calculation of the drift of a Best Estimate of Reactor Power, a weighted average of the Secondary Calorimetric Power Calculation based upon the feedwater venturi meters and the Primary Thermal Power Calculation. The Secondary Calorimetric Power Calculation is used to determine plant power in the event of a loss of LEFM signal. For purposes of calculating drift of the Secondary Calorimetric parameter, one year of data averaged at 10-minute intervals and reported every 15 minutes was evaluated. This allows for potential variability from any seasonal effects. Because the LEFM flowmeters are not yet operating, First Stage pressure was used as the reference against which venturi drift was calculated. First Stage pressure was expected to be stable during the short interval, but any variability of the First Stage pressure indication conservatively adds to the bounding results of the drift calculation.

The analysis established a bounding uncertainty of 0.037% RTP, rounded to 0.04% RTP, over a 7-day period for Oconee Unit 3 at operating levels above 90% RTP. This uncertainty has a 95% statistical probability at a 95% confidence level. The analysis demonstrates that the drift is random and not uni-directional.

Based on this analysis, the venturi meter on which the Secondary Calorimetric is based will be calibrated to the LEFM output plus 0.04%, thereby introducing a 0.04% bias in the venturi reading in the high direction. This will ensure that plant operation based on the Secondary Calorimetric Power Calculation encompasses the additional 0.04% RTP uncertainty indicated by this analysis. Plant power can be kept at the new uprated power level for up to 7 days based upon the Secondary Calorimetric Power Calculation as calibrated to the last acceptable LEFM Calorimetric Power calculation with this additional bias.

If the LEFM signal is not available at the expiration of the 7-day period, the affected Unit will decrease power to the pre-MUR licensed thermal power level. As the calculated uncertainty in RTP was significantly less for Units I and 2, the 0.04% calculated for Unit 3 is a conservative bound for Units 1 and 2 Secondary Calorimetric Calculation drift, which serves as justification of continued operation at the MUR rated thermal power for a 7-day period upon loss of the LEFM signal at these Units 1 and 2, as well.

A Selected Licensee Commitment (SLC) will be added to require the LEFM to be restored in 7 days. If the LEFM is not restored within 7 days, then within six hours the unit will be reduced to no more than 2568 MWt (the previously licensed rated thermal power), the overpower trip setpoint will be reduced, and the flux/flow trip setpoints will be adjusted as specified in the Core Operating Limits Report. If the power level, overpower trip setpoint and flux/flow trip setpoint cannot be reduced within six hours, then the unit shall be placed in Mode 3 within the next six hours.

These requirements ensure that the LEFM inputs are in use whenever power is greater than the pre-uprate RTP level of 2568 MWt and that power will be reduced and maintained at or below the pre-uprate level of 2568 MWt until the LEFM is returned to operable status.

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.

Response to Criterion 2:

Oconee Nuclear Station will not consider a check plus system with a single failure as a separate category; this will be considered as an inoperable LEFM and the same actions identified in response to Criterion 1 above will be implemented.

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.

Response to Criterion 3:

As stated in response to Criterion 2, Oconee Nuclear Station will not consider a check plus system with disabled components as a separate category; this will be considered as an inoperable LEFM and the same actions identified in response to Criterion 1 above will be implemented.

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.

Response to Criterion 4:

The Oconee units have no flow straightener upstream (or downstream) of the LEFM installation and thus this criterion is not applicable to Oconee.

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-754, Revision 0, 'The Effect of the Distribution of the Uncertainty in Steam Moisture Content on the Total Uncertainty in Thermal Power'," March 18, 2010)

Response to Criterion 5:

The Oconee Nuclear Steam Supply Systems (NSSSs) use Once-Through Steam Generators (OTSGs) that produce superheated steam with a very low moisture content. Thus, uncertainty associated with the steam moisture content at Oconee is not a factor in the heat balance uncertainty calculation (See OSC-3737 in Attachment 6). 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 Oconee Unit. The calculations are listed below and are included in Attachment 6 to this LAR. Acceptance testing following installation of the CheckPlus systems in the Oconee 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%.

Parameter	Uncertainty	Power Uncertainty
LEFM Power	0.31 %	0.31 % RTP
Feedwater Pressure	2.13 psi	0.0001 % RTP
Steam Enthalpy		
Temperature	1.25 °F	0.13 % RTP
Pressure	1.26 psi	0.02 % RTP
RCP, Makeup/Letdown Power	0.11 %	0.11 % RTP
Total Secondary Power Uncertainty		0.34 % 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:***

I.1.F.i ***maintaining calibration***

RESPONSE: Calibration of the LEFM will be ensured by preventative maintenance activities previously described in Section I.1.D, Response to Criterion 1.

New instruments that contribute to the power calorimetric will be maintained according to required calibration and maintenance procedures. The other instruments that contribute to the power calorimetric were unaffected by the addition of the LEFM and will be maintained, according to existing calibration and maintenance procedures.

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

RESPONSE: Hardware configuration will be controlled in accordance with Duke Energy procedure, NSD-301, "Engineering Change Program."

LEFM software will be properly classified in accordance with Duke Energy directive NSD-800, "Software and Data Quality Assurance (SDQA) Program" and EDM-801, "Cyber Security Risk Evaluation," and NSD-804, "Cyber Security for Digital Process Systems." Software will be classified, developed, tested, and controlled in accordance with NSD 806, "Digital System Quality Program". Implementation of the software will be performed under the design control process governed by EDM-601, "Engineering Change."

Instruments that affect the power calorimetric, including the Cameron LEFM CheckPlus System inputs, are monitored by Oconee 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

RESPONSE: Corrective actions will be monitored and performed in accordance with Duke Energy procedures NSD 208, "Problem Investigation Process (PIP)" and Work Process Manual.

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

RESPONSE: Reporting deficiencies to the manufacturer will be performed in accordance with Duke Energy directive NSD 208, "Problem Investigation Process (PIP)" and procurement specification.

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

RESPONSE: Manufacturer deficiency reports will be received and addressed in accordance with Duke Energy directive NSD 208, "Problem Investigation Process (PIP)".

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

RESPONSE: The proposed allowed outage time for the instrument and the technical basis for the time selected is provided in the response to I.1.D, 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, criterion 1 from ER-157P above.

References for Section I:

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

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

Ref I.3 Cameron Engineering Report ER-157P, "Supplement to Cameron Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or CheckPlus," Revision 8, May 2008

Ref 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

Ref 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

Ref I.6 Duke Energy Engineering Change Package EC103132, Rev 3, "MUR Power Uprate - Install Cameron LEFM Ultrasonic Flowmeters (2) in ," 2 Dec 2010

Ref I.7 Not used.

Ref I.8 Not used.

Ref I.9 Cameron Engineering Report ER-855, "Meter Factor Calculation and Accuracy Assessments for the LEFM Check Plus Meters at Oconee Units 1, 2 and 3," Revision 0, September, 2010

Ref I.10 Not used

Ref I.11 Cameron Engineering Report ER-813, Rev 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus System," October 2010

Ref I.12 Cameron Engineering Report ER-824 Rev 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus System," October 2010

Ref I.13 Cameron Engineering Report ER-825 Rev 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus System," October 2010

Ref I.14	Duke Energy Calculation OSC-3737, "Secondary Power Uncertainty Analysis," Revision 9, 02 Feb 2011
Ref I.15	Duke Energy Calculation OSC-8856, "Digital RPS Neutron Overpower (Neutron Flux) and Pump Power/Flux Trip Function Uncertainty Analysis," Revision 2, 2 April 2011
Ref I.16	NSD-806, "Digital System Quality Program," Revision 1, 28 March 2011
Ref I.17	NSD-800, "Software and Data Quality Assurance (SDQA) Program," Revision 12, 28 September 2010
Ref I.18	NSD-804, "Cyber Security for Digital Process Systems," Revision 4, 1 June 2011
Ref I.19	ISA-RP 67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation," Approved September 1994
Ref I.20	Duke Energy Directive NSD-208, "Problem Investigation Program (PIP)," Revision 32, 7/29/10
Ref I.21	Duke Energy Directive NSD-301, "Engineering Change Program," Revision 38, 5/31/11
Ref I.22	American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1, "Measurement Uncertainty," 1985

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 – Oconee Analyses. For analyses that remain bounded, all information requested in II.1.A through II.1.D is included in Table II.1. For unbounded analyses, a reference to the applicable portion of Section III is provided.

Each analysis is described briefly below, and all analyses are summarized on Table II-1. The methodology in these analyses is found in Duke Energy Topical Reports, Vendor Topical Reports, and other reports as referenced in Table II.1. NRC review and approval of the applicable report is also referenced in Table II.1.

Reactor Protection System (RPS) 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 uprate is the nuclear overpower (also known as high flux) trip function. In the safety analyses, the trip setpoint is identical to the AV. A summary of key allowable values is tabulated following this discussion.

The high flux trip setpoint AV is currently 105.5% of 2568 MWt (2709.2 MWt). All of the UFSAR safety analyses were revised for the MUR 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% 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 new high flux trip AV for 3 Reactor Coolant Pump (RCP) operation is proposed with this license amendment request. The proposed setpoint maintains the 4 RCP difference between rated thermal power and the high flux trip setpoint, i.e., 5.5% RTP. The nominal 3 RCP operation power level assumed in the safety analyses is unchanged for the MUR 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 setpoint of 79.3% of 2610 MWt, or 2069.7 MWt. The basis for the new setpoint is to better mitigate the UFSAR Chapter 15.17 Small Steam Line Break transient initiated with 3 RCPs in operation. See the small steam line break discussion below.

If the LEFM is out of service for longer than the SLC allowance, the high flux trip setpoint is returned to the pre-MUR uprate value of 2709.2 MWt (103.8% of 2610 MWt). For 3 RCP operation, the setpoint is reduced by 1.7% (1.64% rounded up), from 79.3% to 77.6% of 2610 MWt. This maintains the analytical limit used in the safety analyses upon actuation of the high flux trip.

Comparison of Allowable Values before and after MUR				
Current Allowable Value				MUR Allowable Value
%RTP (2568 MWt)		MWt		%RTP (2610 MWt)
		2754	4 RCP overpower trip setpoint	105.5%
105.5%	4 RCP overpower trip setpoint	2709	4 RCP trip setpoint without LEFM	103.8%
102%	RTP + 2% uncertainty	2619	RTP + 0.34% uncertainty	100.34%
		2610	Rated Thermal Power	100%
100%	Rated Thermal Power	2568		
		2070	3 RCP overpower trip setpoint	79.30%
75%	Max Power for 3 RCP operation	1926	Max Power for 3 RCP operation	73.80%

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 RPS trip functions. Since this accident is initiated from zero power, the MUR 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 adjusted flux related trip setpoints that bound the MUR uprate; and the resulting peak primary pressure is acceptable.

The analysis of record (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.

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 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.

4 Moderator Dilution Accidents (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 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.

5 Cold Water Accident (UFSAR Section 15.5

The cold water accident is analyzed with 3 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 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.

6 Loss of Flow Accidents (UFSAR Section 15.6)

6a Loss of Coolant Flow – Flow Coastdown

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 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 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 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.

6b Loss of Coolant Flow – Locked Rotor

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 departure from nucleate boiling ratio (DNBR) is the acceptance criterion.

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 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.

7 Control Rod Misalignment (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% $\Delta k/k$ 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 uprate, the DNB and CFM results remain acceptable.

The AOR for this classification of accidents are reflected in the Oconee 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.

8 Turbine Trip Accident (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 80% 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 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 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 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.

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 uprate affects the analysis only as it affects the initial stored energy and subsequent cool down. The acceptance criteria 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 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.

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 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 Oconee 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 15.11.2.1)
2. Base Case Fuel Handling Accident Inside Containment (UFSAR 15.11.2.2)
3. Fuel Shipping Cask Drop Accidents (UFSAR 15.11.2.4)
4. Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building (UFSAR 15.11.2.5))
    - Criticality Analyses for Dry Storage Transfer Cask Drop Scenarios
    - Potential Damage to Spent Fuel Pool (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 uprate. The source term and calculated dose results for the fuel handling accidents were reviewed and approved as part of the LAR for full-scope implementation of the Alternative Source Term.

12 Rod Ejection Accident (UFSAR Section 15.12)

The rod ejection accident is analyzed at hot zero power (HZP), at 80% 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, and 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 uprate. Since the 3 RCP cases were initiated from a power level (2054 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 uprate power level, all DNB, peak enthalpy, and peak RCS pressure results are acceptable for a 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.

A separate submittal related to use of Gadolinia as an integral burnable absorber in Oconee cores has been issued (References II.25 and II.26). The Gadolinia LAR has received NRC approval. The Gadolinia LAR revises 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 uprate. Consequently, the method submitted in References II.25 and II.26 and the analyses performed with that method remain valid for the MUR uprate.

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.9. Consequently, the rod ejection dose analysis remains acceptable for the MUR uprate.

13 Steam Line Break Accident (UFSAR Sections 15.13 and 5.2.3.4)

The main steam line break (MSLB) accident described in UFSAR 15.13 is initiated from 102% of 2568 MWt. The MSLB 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 accident is also performed to quantify steam release through the break for input to the dose analysis. The MUR uprate will not affect the DNBR analysis as the initial power level remains bounding. The MUR 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 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 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.

The MSLB 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) Replacement Once Through Steam Generator (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 uprate, the analysis remains acceptable for the MUR 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 (Section 20) below.

14 Loss of Coolant Accidents (UFSAR Section 15.14)

The loss of coolant accidents have been reviewed for the impact of the uprate. Based on the power levels assumed in the current analyses, it has been determined that all of LOCA analyses, bound the uprate. Since the proposed change relies on less than 0.4% uncertainty, the nominal power level of 100.36% 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 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 AREVA using their NRC approved RELAP5 model and was performed at 102% of 2568 MWt. Consequently, the tube stresses following a pressurizer surge line break and hot leg break are acceptable for the MUR 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 (Section 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 LAR for full-scope implementation of the Alternative Source Term (Reference II.9). Consequently, the maximum hypothetical accident radiological analysis remains acceptable for a MUR 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 29 was issued based on analyses performed with the NUREG 0800 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 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 and at 80% of 2568 MWt with 3 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) or the flux/flow trip function (3 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 3 RCP case was initiated from a power level (2054 MWt) that bounds the allowed power level for 3 RCP operation (1926 MWt), and 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 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.

A new high flux trip setpoint for 3 RCP operation is submitted with this LAR. The basis for this new trip setpoint is to better mitigate the 3 RCP small steam line break analysis. The current 3 RCP small steam line break analysis relies on the flux/flow trip function to limit the overpower. 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.

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 uprate.

18 Anticipated Transients Without Scram (ATWS) (UFSAR Section 15.18)

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 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 uprate does not affect the analysis since the initial power level, stored energy of the SSCs, and decay heat are representative of the MUR uprate values for those parameters.

19 Natural Circulation Cooldown (UFSAR Section 5.1.2.4)

Oconee 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 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 uprate does not affect the analysis because decay heat is not explicitly modeled. Either the atmospheric dump valves (ADVs) or 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 uprate.

Furthermore, as discussed in the Appendix R Fire response (Section 24a 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 uprate.

20 Containment Performance (UFSAR 6.2)

Containment short term pressure following a LOCA is discussed in UFSAR 6.2.1.1.3.1; containment long term temperature following a LOCA is discussed in UFSAR 6.2.1.1.3.2; and containment temperature and pressure following a steam line break is discussed in UFSAR 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 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 analyses of record for these analyses are reflected in the Oconee 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.

21 EQ parameters

The pressure and temperature profiles generated by the various transient analyses are summarized here. The MUR 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 all obtain the M&E data from analyses performed at 2619 MWt (102% of 2568 MWt). The containment response analysis following an Appendix R fire or an NFPA-805 based fire (described below in Sections 24a and 24b, respectively) also obtain 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 uprate and result in a conservative pressure/temperature profile for use in the EQ analyses.

Section V.I.C of this Enclosure addresses the environmental qualification of electrical equipment, including the normal and post-accident environmental conditions following the MUR.

22 Standby Shutdown Facility (SSF) Event Turbine Building Flood

A turbine building flood results in the loss of all feedwater as the main and emergency feedwater pumps are in the basement of the turbine building. The reactor will trip on the Anticipatory Reactor Trip System (ARTS) main feedwater pump trip function or on the high RCS pressure trip function. The operators will staff the Standby Shutdown Facility (SSF) to use SSF auxiliary service water (ASW) for long term decay heat removal. The SSF analysis discussed in the Appendix R Fire response (Section 24a below) is the analysis that demonstrates the SSF can successfully maintain the plant at Hot Standby conditions (≥525 °F) for 72 hours following an event that could lead to use of the SSF, such as turbine building floods. As described in the Appendix R fire response below, the analysis was performed at 102% of 2568 MWt (2619 MWt) and successfully demonstrated the plant can be maintained in MODE 3, ≥ 525 °F for 72 hours. Since the SSF analyses are performed assuming an initial power level of 2619 MWt, the analysis results are acceptable relative to the MUR uprate.

There is an interim non-licensing basis analysis for flooding caused by an external flood event. The thermal-hydraulic analysis of this event is performed using the NRC reviewed and approved methods (Reference II.11) and the RETRAN-02 transient analysis code. The analysis is initiated from 2619 MWt. The analysis successfully demonstrates the plant can be cooled to hot shutdown conditions (MODE 4) using normal plant systems and maintained there for a period of 24 hours using the B5b Hale pump and ADVs to flood the generators to keep the RCS in MODE 4. Since the analysis is initiated from a power level that bounds the uprate, the results remain conservative. A long term strategy is being formulated for external flood mitigation and any analyses performed in support of that strategy will account for an MUR uprate.

23 Station Blackout (SBO)

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 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 bound by the load profiles, which are supported by the existing analysis of record. As a result, Standby Shutdown Facility system 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 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, Oconee Nuclear Station is in compliance 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 24 below, Oconee Units 1, 2, and 3 will still be able to achieve safe shutdown following the MUR uprate.

24 Fires

24a Appendix R Fire

The Appendix R thermal-hydraulic fire analysis is performed to demonstrate the SSF can successfully maintain the unit at hot standby (MODE 3) conditions with natural circulation for 72 hours following a fire. Additionally, a separate calculation is performed for a natural circulation cool down from hot standby to hot shutdown (from 560 °F to 200 °F) using SSF ASW, one high pressure injection (HPI) pump (restored after damage mitigation strategies implemented), ADVs, and the reactor vessel head vent. 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 analysis is performed using NRC reviewed and approved methods (Reference II.11). The results demonstrate that natural circulation is successfully established and maintained for both 72 hours at hot standby conditions and for the cool down to 200 °F.

Since the event is analyzed at an initial power level that bounds the MUR uprate power, the Appendix R fire results are acceptable for the MUR uprate.

Currently Oconee is in an operable but degraded/non-conforming (OBDN) condition for this event until either main steam isolation valves (MSIVs) are installed or the transition to NFPA-805 fire analysis occurs. The analysis described in the previous paragraph assumes all steam loads are isolated immediately on turbine trip, which is what would happen if MSIVs were installed. In reality, steam flow paths exist via manually isolated branch lines. An analysis has been performed to demonstrate natural circulation is successfully established and maintained for 72 hours at hot standby conditions assuming the various branch lines are manually isolated within certain time frames. The analysis was initiated from 2619 MWt (102% of 2568) and was also performed using the NRC reviewed and approved methods documented in Reference II.11. Since the OBDN condition was analyzed at an initial power level that bounds the MUR uprate power, and since acceptable results are obtained, the MUR uprate does not affect the OBDN conclusion.

24b - NFPA-805 Fire

NRC approval for transitioning from an Appendix R licensing basis fire analysis to the probabilistic NFPA-805 fire analysis has been obtained contingent upon satisfying the SER requirements (Reference II.34). The due date for completing the SER requirements, as stated in Reference II.34, is January 1, 2013. Consequently, Duke Energy is licensed to the Appendix R Fire analysis until such a time as the SER requirements have been satisfied. 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 uprate.

25 Spent Fuel Pool Accidents (loss of pool cooling)

This is neither a design basis event nor a scoping event for Oconee. It is not a Chapter 15 event; not a natural phenomenon event; the event is not a design basis transient for

Emergency Feedwater (EFW); is not part of Turbine Building flooding scenario. A loss of SFP cooling was postulated to establish heat up rates in order 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. Event only used to assess performance of SFP cooling system.

26 Loss of Main Feedwater (UFSAR Section 10.4.7.3.1)

The loss of main feedwater event (LOMFW) is performed to demonstrate the adequacy of the EFW system. It is initiated from 102% of 2568 MWt and is analyzed for peak RCS pressure. The MUR 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 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.

27 Main Feedwater Line Break (UFSAR 10.4.7.3.2.2 and 5.2.3.4)

UFSAR Section 10.4.7.3.2.2 summarizes the EFW system response following a MSLB or feedwater line break (FWLB). The acceptance criterion listed in the UFSAR for both of these events is 10 CFR 100 dose limits. Since MSLB is more limiting than FWLB with respect to offsite dose release, no dose analysis is performed for the FWLB analysis.

However, FWLB is analyzed to ensure excessive compressive tube forces are precluded as summarized in UFSAR Section 5.2.3.4. Maintaining steam generator tube integrity ensures the MSLB dose results remain limiting. One case demonstrates the adequacy of HPI forced cooling mode of heat transfer. A second case verifies the time critical operator action time of restoring feedwater to the intact steam generator. The acceptance criterion for both cases is that the steam generator tube compressive forces remain below the compressive forces calculated by BWC for the tornado event (Section 31 below). Both cases are initiated from 2619 MWt (102% of 2568). The FWLB occurs downstream of the check valves on one steam generator resulting in the loss of all main feedwater to both generators. A single failure of the intact steam generator EFW control valve coupled with the actuation of the Automatic Feedwater Isolation System (AFIS) results in a loss of all feedwater to both generators. The primary system and steam generator tubes heat up placing compressive forces on the tubes. The MUR uprate does not affect the analyses since the initial power level, stored energy of the SSCs, and decay heat values are based on operation at 102% of 2568 MWt. The analyses are analyzed using the NRC reviewed and approved methods (Reference II.11). Since acceptable results are obtained, the analyses remain acceptable for an MUR power uprate.

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 Section 21 above.

28 LTOP

Low-Temperature Overpressure Protection is discussed in Section IV.4.C.iv.

29 HELB

This accident is not bounded by the existing analysis and is thus discussed in Section III below.

30 RPS/ES Instrument Uncertainties

Uncertainty Calculation:

The RPS and Engineered Safeguards (ES) instrument uncertainty calculations are performed per Reference II.35. The MUR uprate is accomplished by reducing the uncertainty in the secondary side heat balance. As a result of the MUR uprate, full power is increased and consequently any component expressed as a percent of full power span is also potentially impacted. Therefore, the MUR 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 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 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.

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 uprate.

Uncertainty Application:

The NRC approved safety analysis setpoint method documented in Chapter 4, 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 uprate does not affect the Reference II.1 method but does reduce the magnitude of the heat balance uncertainty.

Per Chapter 4, 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. Similar to the high flux trip function, the MUR 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 Oconee. 1, tornado, wind, hurricane; 2, seismic; 3, external floods; 4, snow and ice.

A discussion of each event is presented below:

1 Tornado, wind and hurricane.

The tornado, wind, and hurricane analysis establishes design criteria for SSCs. There is no design analysis or any mitigation calculation on the docket. A design objective stated in UFSAR is to have capability to safely shutdown all three units. The means by which this is to be accomplished and with what SSCs is defined within the docket.

Current Thermal-Hydraulic Analysis:

The current tornado thermal-hydraulic analysis is performed to ensure steam generator tube integrity is maintained. The analysis is performed to demonstrate the steam generator tubes will not fail due to excessive compressive forces as described in UFSAR Section 5.2.3.4. The analysis is initiated from 102% of 2568 MWt (2619 MWt) and was performed using the methods and models documented in Reference II.11. Since the analysis was performed at a power level that bounds the proposed MUR uprate power level, the results remain acceptable following the MUR uprate.

Future Licensing Basis Thermal-Hydraulic Analysis:

Duke Energy submitted a LAR related to Tornado protection in Reference II.36. The LAR is supplemented by responses to additional information requests, of which Reference II.37 includes a response specific to the proposed future licensing basis thermal-hydraulic analysis. The proposed analysis assumes the tornado damages the turbine building and both main steam lines downstream of the location of the proposed MSIVs. It also assumes the 4160V switchgear (which supplies power to the HPI pumps, MFW pumps, and motor driven EFW pumps) is lost. The MSIVs bottle both generators up and keep secondary pressures and temperatures elevated. The SSF is credited with providing ASW for decay heat removal and for supplying reactor coolant makeup via the SSF RC makeup pump. The analysis submitted in Reference II.37 was also submitted as part of the High Energy Line Break LAR (see Section III) and was initiated from 102% of 2568 MWt. The event was analyzed to ensure the SSF could maintain the plant in Mode 3 with $T_{ave} \geq 525$ °F without interruption of single phase natural circulation. NRC approval is still pending, but since the analysis was initiated from a power level that bounds the MUR uprate and natural circulation was successfully demonstrated, the results are acceptable for an MUR uprate

2 Seismic events

Natural phenomena events are not design basis events (DBEs) at Oconee, instead they impose design criteria on SSCs identified for mitigation of accidents. The systems and components to be protected from the effects of natural phenomena are identified within the UFSAR and other licensing documents.

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

Duke Energy submitted information explaining the seismic basis to the NRC in 1994. The NRC addressed that information in Reference II.48. 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.

Thus there is no specific seismic analysis at Oconee, and no specific power level is assumed. The MUR uprate does not affect the current seismic basis for the plant

3 External floods

Natural phenomena events are not DBEs at ONS, instead they impose design criteria on SSCs identified for mitigation of accidents. 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. There is an interim non-licensing basis analysis for flooding caused by an external flood event. The thermal-hydraulic analysis of this event is performed using the NRC reviewed and approved methods (Reference II.11) and the RETRAN-02 transient analysis code. The analysis is initiated from 2619 MWt. The analysis successfully demonstrates the plant can be cooled to hot shutdown conditions (MODE 4) using normal plant systems and maintained there for a period of 24 hours using the B5b Hale pump and ADVs to flood the generators to keep the RCS in MODE 4. Since the analysis is initiated from a power level that bounds the uprate, the results remain conservative. A long term strategy is being formulated for external flood mitigation and any analyses performed in support of that strategy will account for an MUR uprate.

4 Snow and ice

Natural phenomena events are not DBEs at Oconee, instead they impose design criteria on SSCs identified for mitigation of accidents. 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 Oconee, and no specific power level is assumed. The MUR uprate does not affect the current snow and ice basis for the plant.

References for Section II:

II.1 DPC-NE-3005-PA, Revision 3b, "Oconee Nuclear Station UFSAR Chapter 15 Transient Analysis Methodology", July 2009

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)"

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)"

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)"

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 4a, "Thermal-Hydraulic Transient Analysis Methodology", July 2009

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 MDS Report No. OS-73.2, Revision 0, Analysis of Effects Resulting From Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, & 3, April 25, 1973

II.21 MDS Report No. OS-73.2, Supplement 1, Analysis of Effects Resulting From Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, & 3, June 22, 1973

II.22 MDS Report No. OS-73.2, Supplement 2, Analysis of Effects Resulting From Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, & 3, March 12, 1974

II.23 Letter from NRC (R. C. DeYoung) to Duke (A. C. Thies) dated July 26, 1973 (U1 acceptance)

II.24 Safety Evaluation prepared by the Directorate of Licensing related to Oconee Nuclear Station, Units 2 & 3, Docket Nos. 50-270/287, July 6, 1973

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 to the U. S. Nuclear Regulatory Commission from David Baxter, Vice President, Oconee Nuclear Station, Duke Energy Corporation, "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 to Revise Portions of the Updated Final Safety Analysis Report Related to Tornado Licensing Basis; License Amendment Request No. 2006-009," dated June 26, 2008

II.37 Letter to the U. S. Nuclear Regulatory Commission from David Baxter, Vice President, Oconee Nuclear Station, Duke Energy Corporation, "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, Tornado Mitigation License Amendment Request – Response to Request for Additional Information," dated June 24, 2010

II.38 Letter to the U. S. Nuclear Regulatory Commission from David Baxter, Vice President, Oconee Nuclear Station, Duke Energy Corporation, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for HELB events outside of the Containment Buildings; License Amendment Request No. 2008-005," dated June 26, 2008

II.39 Letter to the U. S. Nuclear Regulatory Commission from Dave Baxter, Vice President, Oconee Nuclear Station, Duke Energy Corporation, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for HELB Events outside of the Containment Building -Unit 2; License Amendment Request No. 2008-006," dated December 22, 2008

II.40 Letter to the U. S. Nuclear Regulatory Commission from Dave Baxter, Vice President, Oconee Nuclear Station, Duke Energy Corporation, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Break Events Outside of the Containment Building," License Amendment Request No. 2008-007," dated June 29, 2009

II.41 AREVA NP Proprietary Topical Report BAW-10192P-A, Rev. 0, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants" (Revision 2, August 2008, contains all changes to the original report, but has not yet been approved by the NRC.)

II.42 NRC Letter from J.E. Lyons (NRC) to J. H. Taylor (AREVA NP), Subject: Acceptance for Referencing of Topical Report BAW-1092-P "BWNT Loss-Of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, (TAC No. M89400)," 18 Feb 1997

II.43 NRC Letter from David B. Mathews (NRC) to J. H. Taylor (AREVA NP), Subject: Safety Evaluation of the Babcock & Wilcox Owners Group Submittal Relating to Assumption in the B&W ECCS Analyses (TAC No. M95480), August 1997

II.44 NRC letter from L. A. Wiens (NRC) to J. W. Hampton (Duke Power Company) NRC Safety Evaluation Report (SER) on station blackout, March 10, 1992 (from UFSAR Section 8.3.2.2.4 and OSS-0254.00-00-4005)

II.45 NRC letter from L. A. Wiens (NRC) to J. W. Hampton (Duke Power Company), NRC Supplemental SER on station blackout, December 3, 1992 (from UFSAR Section 8.3.2.2.4 and OSS-0254.00-00-4005)

II.46 AREVA (B&W) Topical Report BAW-10046A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June 1986

II.47 NRC letter from to James H. Taylor (B&W), "Acceptance for Referencing of Licensing Topical Report BAW-10046, Rev. 2 B&W Owners Group Materials Committee "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," 30 April 1986

II.48 Letter dated December 5, 1994 from L. A. Wiens (NRC) to J. W. Hampton (Duke Power Company), addressing Oconee's licensing basis for seismic event, single failure, and LOCA

Table II.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 above	*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 above	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 above	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 above	Reference II.1	References II.2, II.3, II.4, and II.5
(6) 15.6	Loss of Coolant Flow Accidents	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion above	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 above	Reference II.1	References II.2, II.3, II.4, and II.5
(8) 15.8	Turbine Trip Accident	2054 MWt (80% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion above	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 above	Reference II.1	References II.2, II.3, II.4, and II.5

Table II.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
(10) 15.10	Waste Gas Tank Rupture Accident	2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.6	Reference II.6
(11) 15.11	Fuel Handling Accidents	2619 MWt (102% of 2568)	Yes	See discussion above	References II.7 and II.8	Reference II.9
(12) 15.12	Rod Ejection Accident	0 MWt 1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.1 References II.7 and II.8	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 above	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 above	Reference II.41	References II.42, II.43
(15) 15.15	Maximum Hypothetical Accident	2619 MWt (102% of 2568)	Yes	See discussion above	References II.7 and II.8	Reference II.9
(16) 15.16	Post-Accident Hydrogen Control	2619 MWt (102% of 2568)	Yes	See discussion above	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 above	Reference II.1	References II.2, II.3, II.4, and II.5
(18) 15.18, 7.8	Anticipated Transients Without Scram	2619 MWt (102% of 2568)	Yes	See discussion above	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 above	Reference II.16	Reference II.17

Table II.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
(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 above	Reference II.18	References II.4 and II.19
(21)	EQ parameters	2619 MWt (102% of 2568)	Yes	See discussion above	-See Containment Performance -See App. R fire -See NFPA-805 fire -Penetration room – Reference II.11	-See Cont. Performance -See App. R fire -See NFPA-805 fire - Pen. Room – References II.4, II.5, II.12, II.13, II.14 and II.15
(22)	SSF Event Turbine Building Flood (TBF)	2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.11	References II.4, II.5, II.12, II.13, II.14 and II.15
(23)	Station Blackout (SBO)	See discussion above	Yes	See discussion above	Ref II.44	Ref II.45
(24a)	Appendix R Fire	2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.11	References II.4, II.5, II.12, II.13, II.14 and II.15
(24b)	NFPA-805 Fire	2619 MWt (102% of 2568)	Yes	See discussion above	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 above	See discussion above	See discussion above

Table II.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**
(26) 10.4.7.3.1	Loss of Main Feedwater	2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.11	References II.4, II.5, II.12, II.13, II.14 and II.15
(27) 10.4.7.3.2.2, 5.2.3.4	Main Feedwater Line Break	2619 MWt (102% of 2568)	Yes	See discussion above	Reference II.11	References II.4, II.5, II.12, II.13, II.14 and II.15
(28)	LTOP	0	0	See discussion above	Reference II.46	Reference II.47
(29)	High Energy Line Break/Pipe Rupture (HELB)	2568	No	See discussion above	References II.20, II.21 and II.22	References II.23 and II.24
(30)	RPS/ES Instrument Uncertainties	0 MWt	Yes	See discussion above	Reference II.35	Reference II.4
(31)	Natural Phen: Tornado (incl. missiles), Wind, Hurricane	See discussion above	See discussion above	See discussion above	Reference II.11	References II.4, II.5, II.12, II.13, II.14 and II.15

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, Subsections 1 through 18; and Table II.1 items 1 through 18, for discussion of the Oconee UFSAR Chapter 15 accident analyses. Oconee has no UFSAR Chapter 15 analyses that require re-evaluation for the MUR uprate. Duke Energy also reviewed non-UFSAR analyses and identified two analyses that require re-evaluation for the MUR uprate, those analyses are discussed below.

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: Oconee has no reload analyses that require re-evaluation for the MUR uprate. Various reload analyses (See Section II, number 4, moderator dilution accidents, number 6, flow coastdown and number 7, control rod misalignment) will be performed for each fuel cycle in accordance with normal cycle design practice, but there will be no change to those analyses or their methodology based on the MUR 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: Three analyses are not bounding for the MUR uprate. Each re-analysis is discussed separately below.

1. High Energy Line Break

Current Licensing Basis:

The current HELB licensing basis is described in References II.20, II.21, and II.22. Due to the age of the licensing basis report and its supplements, the initial power level assumption is not clearly defined for the various scenarios analyzed. Duke Energy has conservatively assumed that rated thermal power (2568 MWt) was used in the original analyses. In the current licensing basis, EFW injection within 15 minutes and HPI injection within 60 minutes are needed to mitigate the various HELBs described in the report. As there is currently no assurance of restoring or injecting HPI within the required 60 minute time frame, Oconee is considered operable but degraded/non-conforming (OBDN).

Duke Energy has analyzed various MSLBs and FWLBs outside containment using RETRAN-02 and determined that with a 60 minute HPI restoration, FWLB results in the least margin to core uncovery. Additional FWLB analyses were performed using RELAP5/MOD2-B&W at an initial power level of 2619 MWt (102% of 2568 MWt). The NRC approved use of RELAP5 by Duke Energy for SBLOCA and LBLOCA M&E release applications in Reference II.18. The results demonstrated that decay heat can be successfully removed and the core remains covered for HPI restoration times of up to eight hours. Since the RELAP5 analysis was performed at a power level that bounds the uprated power, the OBDN conclusion remains valid for the MUR uprate.

Reconstituted HELB licensing basis:

Oconee is in the process of reconstituting its HELB licensing basis to resolve the OBDN condition. Three revised HELB analyses (main steam line break, feedwater line break, and letdown line break) were prepared and submitted to the NRC in References II.38, II.39 and II.40. These analyses take credit for the proposed Protected Service Water (PSW) System and the SSF to achieve a safe shutdown condition following certain HELBs postulated throughout the plant. Additionally, proposed MSIVs are necessary to achieve safe shutdown following certain MSLB when using the SSF for event mitigation. These analyses will be the future licensing basis once the NRC SER is received and the PSW system is installed (tentatively scheduled for middle of 2012) and the MSIVs are installed. The MSIVs are currently scheduled to be installed on ONS Units 1, 2 and 3 in 2014, 2015, & 2016, respectively.

The limiting MSLB DNBR analysis submitted via Reference II.38 is a steam line break with coincident LOOP initiated from 2568 MWt. It does not credit MSIV closure to limit the primary system depressurization. Following the submittal of Reference II.40, the MSLB analysis was revised to initiate from 2619 MWt (102% of 2568 MWt). The conclusion in Reference II.40 that DNBR remains within acceptable limits remains valid for the analysis initiated from 2619 MWt. Since the analysis was performed at power level that bounds the MUR uprate, the DNBR results are acceptable for the MUR uprate. The other MSLB cases submitted were all initiated from 2619 MWt and all successfully demonstrate safe shutdown can be achieved crediting either MSIVs or the PSW system. Consequently, these analyses remain acceptable for the MUR uprate.

The FWLB analyses submitted were either performed at 2619 MWt or qualitatively described relative to actuation of mitigating systems to attain safe shutdown conditions and, ultimately, Cold Shutdown (MODE 5) using analyses based on 2619 MWt as the basis for the plant response. Since safe shutdown is demonstrated for a power level that bounds the MUR uprate, the analysis results remain acceptable for the MUR uprate.

The letdown line break analysis submitted was initiated from 2619 MWt to supply input to a dose analysis. The letdown line break analysis does not rely on either the SSF or PSW for successful mitigation. The radiological dose analysis for the letdown line break analysis 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.9. Consequently, the letdown line break dose results remain acceptable for the MUR uprate.

2. Double steam line break

Duke Energy previously performed a plant response analysis of a Double Main Steam Line Break to evaluate the performance of the Oconee Safe Shutdown Facility (SSF) at the 2568 MWt power level. The event includes a loss of offsite power (LOOP) SSF Standby Power supplying power to SSF Auxiliary Service Water (ASW) System, SSF Reactor Coolant Makeup (RCM) System, and pressurizer heaters.

Consistent with the previous analysis, the event is initiated on a best estimate basis not considering power measurement uncertainty. The previous analysis, however was evaluated with the computer code (TRAP2) and thus is being re-performed with currently approved computer code RELAP5/MOD2-B&W. The NRC approved the use of this code (BAW-10164) by Reference III.1. The event is evaluated on the ability of the SSF to mitigate the event such that (i) the core will not return to criticality, (ii) the active fuel will not be uncovered, and (iii) long-term natural circulation will not be halted. Operator actions are credited to manually initiate SSF ASW at 14 minutes, to initiate SSF RCM at 20 minutes, and to open the SSF letdown line when conditions warrant.

Preliminary analysis confirms that the results are acceptable. Improvements to the previous analysis that are being incorporated into the new analysis include the tripped rod worth, the timing and quantity of feedwater flow, the steam generator model, and the timing and quantity of reactor coolant makeup.

Consequently, it is expected that the event analyzed at MUR conditions will provide acceptable results. The Duke Energy commitment to complete this analysis is in Attachment 1 to this LAR.

References for Section III:

III.1 NRC Letter (Ho K. Nief) to AREVA (R. Gardner), June 25, 2007, ML071620460, "Final Safety Evaluation for AREVA NP, INC. Topical Report (TR) BAW-10164(P), Revision 6, RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA [Loss-of-Coolant Accident] and Non-LOCA Transient Analysis (TAC No. MD2187)."

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 critical primary system parameters.

Table IV-1: Critical Primary System Parameters

Parameter	100% Current Licensed Power With 10% SG Tube Plugging	102% Current Licensed Power With 10% SG Tube Plugging
Core Thermal Power (MWt)	2568	2619
Other RCS Power MWt (RCP Heat)	16	16
Total RCS Thermal Power (MWt)	2584	2635
RCS Pressure (psig)	2155	2155
T_{hot} (°F)	601.7	602.1
T_{cold} (°F)	556.4	556.0
T_{ave} (°F)	579.1	579.1
RCS Mass Flow (E6 Lbm/hr)	145.5	145.52

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

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

RESPONSE: The MUR uprate conditions were reviewed for impact on the design basis analyses for the reactor vessel (including vessel nozzles). No changes to the RCS 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 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. Therefore, the existing stress reports for the reactor vessel remain applicable for the power uprate conditions.

Reactor vessel supports are included with the reactor coolant system supports discussed in Section IV.1.A.iv below.

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

RESPONSE: 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 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 effects of operating temperature changes (T_{hot} and T_{cold}) are within design limits. 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 Oconee 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 Oconee 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 uprate.

Duke Energy participates in industry activities associated with the development of the standard industry guidance, including activities performed by the PWR Owners Group (PWROG) and EPRI PWR Materials Reliability Program (MRP). In 2008, EPRI issued the MRP-227-Rev. 0, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," which provides generic inspection and evaluation guidelines based on a broad set of assumptions about plant operation. The Oconee reactor internals inspection plan (ANP-2951, Rev. 1) was developed in accordance with the generic requirements in MRP-227, Rev. 0.

Duke Energy assessed the impact of the proposed MUR power uprate on the Oconee reactor internals inspection plan. The assessment addressed the impact of MUR uprate on aging degradation in reactor internals including the following aging degradation concerns identified in NRC RS-001.

- Irradiation-assisted stress corrosion cracking
- Thermal and neutron embrittlement of cast austenitic stainless steel
- Stress corrosion cracking
- Void swelling

The assessment examined the effect of increased fluence and changes in the coolant temperature, and concluded that the Oconee reactor vessel internals inspection plan will not be affected by the MUR power uprate.

In June 22, 2011, NRC issued the final safety review of the MRP-227-Rev. 0. The NRC approved version of MRP-227-A is projected to be issued by the end of 2011 by EPRI. Duke Energy remains committed to incorporate recommendations from MRP programs that are applicable to the three Oconee units. As part of this ongoing commitment, Duke Energy will review MRP-227-A, and, if needed, revise the Oconee reactor vessel internals inspection plan. In addition, Duke Energy will continue to participate in industry activities related to aging issues in PWR reactor internals including the activities related to MRP-227.

IV.1.A.iii control rod drive mechanisms

RESPONSE: The MUR uprate conditions were reviewed for impact on the existing design basis analyses for the control rod drive mechanisms. 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 control rod drive mechanisms remain applicable for the power uprate conditions.

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

RESPONSE: The MUR uprate conditions were reviewed for impact on the existing design basis analyses for the reactor coolant piping and supports. 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 piping and supports remain applicable for the power uprate conditions.

There is a discussion of thermal stratification, and Bulletin 88-01, 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)

RESPONSE: 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's 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 uprate.

The 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 functions 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 Reactor Coolant Makeup (RCM) System 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 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

RESPONSE: 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 U.S. NRC Regulatory Guide 1.121 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

RESPONSE: The MUR 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

RESPONSE: The MUR 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 of the pressurizer surge line, and Bulletin 88-01, in Section IV.1.B.iv.

IV.1.A.ix safety-related valves

RESPONSE: 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 DBE analysis and thus there is no adverse impact on the pressurizer overpressure protection valves from the MUR power uprate.

Other safety-related valves were reviewed as part of the system that contains those valves.

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

IV.1.B.i stresses

RESPONSE: Since the MUR power uprate conditions are bounded by the conditions evaluated for 102% (2619 MWt) of the current Rated Thermal Power; the MUR power uprate conditions have a negligible effect on the RCS structural analysis. The design conditions used in the existing analyses remain bounding for the MUR power uprate.

For the increase in power from 2568 MWt to 2619 MWt (which bounds the MUR 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.

The MUR uprate conditions were reviewed for impact on the existing design basis analyses. 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 and stresses remain valid.

IV.1.B.ii cumulative usage factors

RESPONSE: Since the MUR power uprate conditions are bounded by the conditions evaluated for 102% (2619 MWt) of the current Rated Thermal Power; the MUR power uprate conditions have a negligible effect on the RCS structural analysis. The design conditions used in the existing analyses remain bounding for the MUR power uprate.

The revised design conditions were reviewed for impact on the existing design basis analyses. 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 (CUFs) remain valid.

There is a discussion of thermal stratification, and Bulletin 88-01, in Section IV.1.B.iv.

IV.1.B.iii flow induced vibration

RESPONSE: For the increase in power from 2568 MWt to 2619 MWt (which bounds MUR 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.

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

RESPONSE: Thermal stratification in the lines attached to the RCS occurs mainly during heatup and cooldown. The 100% power hot and cold leg temperatures that the plant has been designed to bound the temperatures for the MUR uprate. Therefore, the effects of thermal stratification will be the same or less as a result of the power uprate.

NRC Bulletin 88-08 addresses the issue of thermal stresses in piping attached to the primary loop that cannot be isolated. The operating temperature changes as a result of the MUR uprate compared to the current operating temperatures are negligible and will not have an effect on the existing or potential thermal stratification conditions. In addition, the design RCS flow rates are essentially the same as those for the MUR uprate and therefore, the effects of the turbulence around penetrations will not change as a result of power uprate.

NRC Bulletin 88-11 addresses the issue of surge line thermal stratification. Thermal stratification in the surge line occurs mainly during 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 due to the MUR uprate. An increased hot leg temperature gives a lower temperature differential between the hot leg and the pressurizer which in turn lessens the stratification effects. This indicates 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 analyses.

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

RESPONSE: The system design pressure remains unchanged.

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

RESPONSE: There is no change in design RCS flow for the MUR 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 Sections II and III.

IV.1.B.vii high-energy line break locations

RESPONSE: The impact of the MUR uprate on HELB locations inside and outside containment at ONS were evaluated. As discussed in Sections IV.1.B.i and IV.B.1.ii, the design conditions are not changing for the MUR, 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 III of this enclosure.

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

RESPONSE: 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 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 Tave 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 uprate design conditions were reviewed for impact on the existing hydraulic forcing functions. It is 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:

IV.1.C.i pressurized thermal shock calculations

RESPONSE:

The reference temperature for pressurized thermal shock (RT_{PTS}) for the reactor vessel beltline materials was reevaluated for the MUR power uprate at a projected end-of-life of 48 EFPY. RT_{PTS} values were calculated in accordance with the requirements in 10 CFR 50.61. The neutron fluence used for this evaluation is discussed in Section IV.1.C.ii below.

The limiting reactor vessel beltline material for Oconee Unit 1 has a RT_{PTS} value of 229.1°F at 48 EFPY; the screening criterion for this material is 270°F.
The limiting reactor vessel beltline material for Oconee Unit 2 has a RT_{PTS} value of 296.3°F at 48 EFPY; the screening criterion for this material is 300°F.
The limiting reactor vessel beltline material for Oconee Unit 3 has a RTPTS value of 251.4°F at 48 EFPY; the screening criterion for this material is 300°F.

The calculations for these values are shown in Table IV.1.C-1. The Oconee reactor vessels will remain within their limits for RT_{PTS} after the MUR power uprate.

Surveillance Data: Consideration of Oconee plant-specific surveillance information was considered in the projection of RT_{PTS}. Plant-specific CF values from surveillance information are used for calculating ΔRT_{NDT} for the following two materials:

- ANK 191, lower shell forging for Oconee Unit 3
- AWS 192, upper shell forging for Oconee Unit 3

The Chemistry Factors for all other materials in this document are determined using Table 1 (for weld metals) and Table 2 (for base metals) of 10 CFR 50.61 without correction.

Table IV.1.C-1 RT_{PTS} projections for the MUR Uprate at Oconee Nuclear Station

Oconee Unit 1

Material Description							Chemistry			48 EFPY Values					
Reactor Vessel Beltline Region Material	Material Identifier	Heat Number	Type	Initial RT_{NDT} (°F)	σ_i	σ_Δ	Cu wt%	Ni wt%	Chem. Factor (°F)	Fluence 0T (n/cm^2)	Fluence Factor	ΔRT_{PTS} (°F)	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
Lower Nozzle Belt (LNB) Forging	AHR-54	ZV-2861	A-508, Cl. 2	3	31.0	17.0	0.16	0.65	119.3	1.08E+18	0.43	51.6	70.7	125.3	270
Intermediate Shell (IS) Plate	C2197-2	C2197-2	SA-302, Gr. B	1	26.9	17.0	0.15	0.5	104.5	1.15E+19	1.04	108.6	63.6	173.2	270
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302, Gr. B	1	26.9	17.0	0.1	0.5	65.0	1.28E+19	1.07	69.5	63.6	134.1	270
Upper Shell Plate	C3278-1	C3278-1	SA-302, Gr. B	1	26.9	17.0	0.12	0.6	83.0	1.28E+19	1.07	88.7	63.6	153.3	270
Lower Shell (LS) Plate	C2800-1	C2800-1	SA-302, Gr. B	1	26.9	17.0	0.11	0.63	74.5	1.25E+19	1.06	79.1	63.6	143.7	270
Lower Shell Plate	C2800-2	C2800-2	SA-302, Gr. B	1	26.9	17.0	0.11	0.63	74.5	1.25E+19	1.06	79.1	63.6	143.7	270
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	-5	19.7	28.0	0.23	0.52	157.4	1.08E+18	0.43	68.1	68.5	131.6	300
IS Long. Weld (100%)	SA-1073	1P0962	Linde 80	-5	19.7	28.0	0.21	0.64	170.6	9.00E+18	0.97	165.6	68.5	229.1	270
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	10	0.0	28.0	0.23	0.59	167.6	1.15E+19	1.04	174.1	56.0	240.1	300
US Long. Weld (100%)	SA-1493	8T1762	Linde 80	-5	19.7	28.0	0.19	0.57	152.4	1.15E+19	1.04	158.3	68.5	221.8	270
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	-5	19.7	28.0	0.22	0.54	158.0	1.22E+19	1.06	166.8	68.5	230.3	300
LS Long. Weld (100%)	SA-1426	8T1762	Linde 80	-5	19.7	28.0	0.19	0.57	152.4	1.04E+19	1.01	154.0	68.5	217.5	270
LS Long. Weld (100%)	SA-1430	8T1762	Linde 80	-5	19.7	28.0	0.19	0.57	152.4	1.04E+19	1.01	154.0	68.5	217.5	270

Oconee Unit 2

Material Description							Chemistry			48 EFPY Values					
Reactor Vessel Beltline Region Material	Material Identifier	Heat Number	Type	Initial RT_{NDT} (°F)	σ_i	σ_Δ	Cu wt%	Ni wt%	Chem. Factor (°F)	Fluence 0T (n/cm^2)	Fluence Factor	ΔRT_{PTS} (°F)	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
LNB Forging	AMX 77	123T382	A-508, Cl. 2	3	31.0	17.0	0.13	0.76	95.0	1.14E19	1.04	98.5	70.7	172.2	270
US Forging	AAW 163	3P2359	A-508, Cl. 2	20	0.0	17.0	0.04	0.75	26.0	1.28E19	1.07	27.8	27.8	75.6	270
LS Forging	AWG 164	4P1885	A-508, Cl. 2	20	0.0	17.0	0.02	0.8	20.0	1.27E19	1.07	21.3	21.3	62.6	270
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	-5	19.7	28.0	0.27	0.59	182.6	1.14E19	1.04	189.2	68.5	252.7	300
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	-7	20.6	28.0	0.34	0.68	220.6	1.24E19	1.06	233.8	69.5	296.3	300

Oconee Unit 3

Material Description							Chemistry			48 EFPY Values					
Reactor Vessel Beltline Region Material	Material Identifier	Heat Number	Type	Initial RT_{NDT} (°F)	σ_i	σ_Δ	Cu wt%	Ni wt%	Chem. Factor (°F)	Fluence 0T (n/cm^2)	Fluence Factor	ΔRT_{PTS} (°F)	Margin (°F)	RT_{PTS} (°F)	Screening Criteria (°F)
LNB Forging	4680	4680	A-508, Cl. 2	3	31.0	17.0	0.13	0.91	96.0	1.10E+19	1.03	98.6	70.7	172.3	270
US Forging	AWS 192	522314	A-508, Cl. 2	40	0.0	17.0	0.01	0.73	36.0*	1.22E+19	1.06	38.0	34.0	112.0	270
LS Forging	ANK 191	522194	A-508, Cl. 2	40	0.0	8.5	0.02	0.76	17.4*	1.21E+19	1.05	18.3	17.0	75.3	270
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	-5	19.7	28.0	0.24	0.63	178.0	1.10E+19	1.03	182.7	68.5	246.2	300
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	-5	19.7	28.0	0.26	0.60	180.0	1.17E+19	1.04	187.9	68.5	251.4	300

* - Chemistry factor adjusted based on surveillance data.

IV.1.C.ii fluence evaluation

RESPONSE:

For consistency with the ONS license renewal application, the embrittlement study for the MUR power uprate was conducted for an end of life of 48 EFPY. The NRC approval of the Oconee License Renewal Application based on 48 EFPY is found in NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," March, 2000.

The MUR fluence values were determined by taking the most recently calculated vessel fluence value for each ONS unit vessel and adding the estimated fluence obtained using the projected fluence rates (pre-MUR and MUR) and time periods out to the 40 year (33 EFPY) and 60 year (48 EFPY) vessel life. The most recent fluence (E > 1.0 MeV) values were calculated using core follow information and the NRC approved, Regulatory Guide 1.190 compliant, BAW-2241P-A methodology. The pre-MUR fluence rate used in the projection was selected as the highest fluence rate from the most recent 5 (unit 1) or 6 (units 2 and 3) cycles. The MUR fluence rate was determined by increasing the pre-MUR fluence rate by 2 percent. The pre-MUR time period used was the cycle(s) after the last full fluence transport analysis was performed for the cycle the MUR is assumed to start in (cycles 27 for unit 1, cycle 25 for unit 2, and cycle 26 for unit 3). The MUR time period used was from the start of the next cycle (cycle 28 for unit 1, cycle 26 for unit 2, and cycle 27 for unit 3), until the end of current and extended vessel life, 40 years (33 EFPY) and 60 years (48 EFPY).

> BAW-2241 was approved by NRC letter from Frank Akstulewicz to J.J. Kelly (B&W Owner's Group), "Acceptance for Referencing of Licensing Topical Report BAW-2241-P, "Fluence and Uncertainty Methodologies" (TAC NO. M98962)," undated

Fluence values at the clad to base metal interface for all Oconee beltline materials are given in Table IV.1.C-1; while 1/4T fluence values for all Oconee beltline materials are given in Table IV.1.C-2.

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

RESPONSE: The current Technical Specification Pressure-Temperature (P-T) limit curves are licensed through 33 EFPY and are based on adjusted reference temperatures (ART) at the ¼ thickness (¼T) and ¾ thickness (¾T) wall locations for the limiting reactor vessel beltline material. The impact of the MUR power uprate on the P-T curves was assessed by performing revised 33 EFPY ART calculations in accordance with Regulatory Guide 1.99, Revision 2, which considered reactor vessel surveillance data and the post-MUR uprate fluence as described in Section IV.1C.ii above. Based on these fluence projections, the limiting ART values post-MUR power uprate are bounded by the corresponding pre-MUR power uprâte ART values except for one beltline region base metal ART value. For that base metal, the ¾T ART value increased by only 0.1°F. The ¾T ART value for this base metal was determined to be not controlling in the development of the P-T limit curves. Therefore, the existing 33 EFPY P-T curves established in accordance with the requirements of 10 CFR Part 50, Appendix G and the associated Lower Temperature Overpressure Protection (LTOP) limits remain valid for the MUR power uprate.

IV.1.C.iv low-temperature overpressure protection

RESPONSE: Since existing 33 EFPY pressure-temperature limit curves remain unaffected by the MUR, the LTOP Limit also remains unaffected.

IV.1.C.v upper shelf energy

RESPONSE: Upper-shelf energy values were calculated using the fluence projections from Section IV.1.C.ii above for the MUR power uprate at 48 EFPY to ensure compliance with 10 CFR Part 50 Appendix G. If the limiting reactor vessel beltline material's Charpy upper-shelf energy (USE) is projected to fall below 50 ft-lb, an equivalent margins analysis must be performed. The reactor vessel beltline base materials at the three Oconee units have projected USE values that are above 50 ft-lb. The reactor vessel beltline weld materials have projected USE values that fall below 50 ft-lb except for one weld at Oconee Unit 1 which is projected to be above 50 ft-lb. These values are shown in Table IV.1.C-2.

An equivalent margins analysis assessment compared the projected fluence values at the inside surfaces as well as at the 1/4T depth locations of these welds for both the pre-MUR condition and for the MUR power uprate condition. The assessment demonstrated that the fluence used for the pre-MUR evaluation at each of these weld locations bounded the fluence projections with the MUR power uprate. Therefore, it is concluded that the current equivalent margins analyses remain valid and applicable for the Oconee units even after the MUR power uprate.

Table IV.1.C-2 - Upper Shelf Energy post-MUR

Oconee Unit 1

Material Description							48 EFPY - Post MUR		
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Location	Initial USE, ft-lb	Fluence 10^{19} n/cm^2	% Drop in USE	USE ft-lb
Lower Nozzle Belt (LNB) Forging	AHR-54	ZV-2861	A-508, Cl. 2	0.16	1/4T	109	1E+18	14.5	93.2
Intermediate Shell (IS) Plate	C2197-2	C2197-2	SA-302, Gr. B	0.15	1/4T	81	6.78E+18	21.9	63.3
Upper Shell (US) Plate	C3265-1	C3265-1	SA-302, Gr. B	0.10	1/4T	81	7.54E+18	17.8	66.6
Upper Shell Plate	C3278-1	C3278-1	SA-302, Gr. B	0.12	1/4T	81	7.54E+18	19.6	65.1
Lower Shell (LS) Plate	C2800-1	C2800-1	SA-302, Gr. B	0.11	1/4T	81	7.37E+18	18.6	65.9
Lower Shell Plate	C2800-2	C2800-2	SA-302, Gr. B	0.11	1/4T	81	7.37E+18	18.6	65.9
LNB to IS Circ. Weld (100%)	SA-1135	61782	Linde 80	0.23	1/4T	70	1E+18	21.4	55.0
IS Long. Weld (100%)	SA-1073	1P0962	Linde 80	0.21	1/4T	70	5.33E+18	30.2	48.9
IS to US Circ. Weld (ID 61%)	SA-1229	71249	Linde 80	0.23	1/4T	70	6.78E+18	33.8	46.4
IS to US Circ. Weld (OD 39%)	WF-25	299L44	Linde 80	0.34	3/4T	70	2.46E+18	34.3	46.0
US Long. Weld (100%)	SA-1493	8T1762	Linde 80	0.19	1/4T	70	6.84E+18	30.2	48.9
US to LS Circ. Weld (100%)	SA-1585	72445	Linde 80	0.22	1/4T	70	7.25E+18	33.4	46.6
LS Long. Weld (100%)	SA-1426	8T1762	Linde 80	0.19	1/4T	70	6.14E+18	29.4	49.4
LS Long. Weld (100%)	SA-1430	8T1762	Linde 80	0.19	1/4T	70	6.14E+18	29.4	49.4

Oconee Unit 2

Material Description							48 EFPY - Post MUR		
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Location	Initial USE, ft-lb	Fluence 10^{19} n/cm^2	% Drop in USE	USE ft-lb
LNB Forging	AMX 77	123T382	A-508, Cl. 2	0.13	1/4T	109	6.78E+18	20.1	87.1
US Forging	AAW 163	3P2359	A-508, Cl. 2	0.04	1/4T	128	7.54E+18	12.2	112.4
LS Forging	AWG 164	4P1885	A-508, Cl. 2	0.02	1/4T	140	7.49E+18	10.3	125.6
LNB to US Circ. Weld (100%)	WF-154	406L44	Linde 80	0.27	1/4T	70	6.78E+18	37.4	43.8
US to LS Circ. Weld (100%)	WF-25	299L44	Linde 80	0.34	1/4T	70	7.31E+18	40.4	41.7

Oconee Unit 3

Material Description							48 EFPY - Post MUR		
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Location	Initial USE, ft-lb	Fluence 10^{19} n/cm^2	% Drop in USE	USE ft-lb
LNB Forging	4680	4680	A-508, Cl. 2	0.13	1/4T	109	6.55E+18	19.9	87.3
US Forging	AWS 192	522314	A-508, Cl. 2	0.01	1/4T	90	7.25E+18	9.3	81.7
LS Forging	ANK 191	522194	A-508, Cl. 2	0.02	1/4T	110	7.14E+18	10.2	98.8
LNB to US Circ. Weld (100%)	WF-200	821T44	Linde 80	0.24	1/4T	70	6.55E+18	34.4	45.9
US to LS Circ. Weld (ID 75%)	WF-67	72442	Linde 80	0.26	1/4T	70	6.96E+18	36.7	44.3
US to LS Circ. Weld (OD 25%)	WF-70	72105	Linde 80	0.32	3/4T	70	2.53E+18	33.2	46.7

IV.1.C.vi surveillance capsule withdrawal schedule

RESPONSE:

All surveillance capsules have been removed from the Oconee Units 1, 2, and 3 reactor vessels. The Oconee reactors are included in the Integrated Surveillance Program, the "The Master Integrated Reactor Vessel Surveillance Program" as described in BAW report 1543, Revision 4, Supplement 6-A. All of the capsules for which the Oconee units take credit at 48 EFPY after MUR power uprate have been withdrawn. Based on the projected fluence values discussed in Section 4.2.3.2 and a review of BAW-1543, Revision 4, Supplement 6-A, the surveillance capsule withdrawal schedule meets the requirements of 10 CFR 50 Appendix H.

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

RESPONSE:

The impact of the MUR power uprate on the previous reactor vessel underclad cracking 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 Oconee units. The underclad cracking analysis is performed for three regions of the reactor vessel: the beltline region, the nozzle belt region and the RV closure head/RV flange region. For each of the three regions it was determined that the limiting adjusted reference temperature (ART) at the inside surface (currently referred to as RT_{PTS}) of the Oconee reactor vessels is less than that used in the previous generic analysis. At 48 EFPY with MUR, the limiting Oconee RT_{PTS} value for the nozzle belt region is 172.3 °F (Unit 3 nozzle belt forging), whereas the RT_{PTS} value used in the previous analysis was 175 °F.

Also, since the previous analysis was performed, the steam generators of the Oconee units have been replaced. Hence, the new Oconee loads and associated stresses on the reactor vessels also had to be determined and compared against the loads/stresses used in the previous analysis. The new Oconee specific loads/stresses were determined to be approximately 50% lower than the conservative generic loads used in the previous analysis.

The results of this underclad cracking assessment demonstrated that the previous underclad cracking analysis performed for 48 EFPY without MUR remains valid and applicable for 48 EFPY with MUR for the Oconee units

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.

RESPONSE: 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.

UFSAR Table 5-4, Reactor Coolant System Component Codes

Component	Codes	Addendum
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 19671
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.

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:

Inservice Inspection:

Duke Energy has reviewed the Inservice Inspection (ISI) Program relative to the MUR uprate. Since the LEFM will not fall under the ISI Program and tested/inspected components will not change as a result of the MUR, there is no change to the ISI program. Therefore, the ISI program is bounded by existing analyses and no changes are required.

Inservice Testing

Duke Energy has reviewed the Inservice Testing (IST) Program relative to the MUR uprate. Since the LEFM will not fall under the IST Program and there are no changes to systems or components (excluding the LEFM) as a result of the MUR, there is no change to the IST Program. Therefore, the IST program is bounded by current analysis and no changes are required.

Flow Accelerated Corrosion

The MUR power uprate will not have a significant impact on the Flow Accelerated Corrosion (FAC) Program. Performing the MUR will impact the FAC related piping wear rates (thinning of pipe walls); however, the changes will be minimal. The short-term effect on the FAC program will be the need to update the existing CHECWORKS modeling software with the new operating parameters (flowrate, temperature, etc.), determine if any formerly non-susceptible piping has become susceptible due to temperature increases, and review the projected future inspection dates to determine if components may need to be inspected/replaced earlier than previously planned. The longer term impact includes performing inspections and replacements of some components earlier than previously projected based on the historical and current plant operating conditions. The impact on the future piping wear rates will be determined through the use of the CHECWORKS modeling software. It is expected that the feedwater system will experience the largest increase in wear. However, it should be noted that, even in the feedwater lines, the wear rate changes caused by the MUR may be undetectable using measurement techniques. This is because velocity changes are predicted to be minimal, thereby causing little change in the wear rates experienced by the systems.

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.

After entering service the Oconee ROTSGs experienced tube wear greater than expected based on the original analyses because of axial flow induced vibration. An experimental program has concluded that the MUR uprate up will have an insignificant effect on axial flow induced vibration and consequently will have an insignificant effect on ROTSG tube wear rate, tube plugging and life expectancy.

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.

The increase in bundle entrance velocity due to the MUR conditions will be 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 Oconee 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 Oconee ROTSGs on the basis that:

- the FIV analysis demonstrated an acceptable fluid-elastic instability ratio for Oconee 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 Oconee 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 AREVA document 51-9156836-002, "NSSS Evaluations for Oconee Units 1, 2, & 3 MUR," July 2011

IV.2 AREVA document 51-9154550-001, "Oconee MUR Power Uprate BOP System Review," July 2011

IV.3 AREVA Topical Report No. BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration," September 1972

IV.4 AREVA Document 32-9156294-000, "Oconee Unit 1, 2 and 3 MUR Power Uprate Evaluation, RV Internals FIV."

IV.5 AREVA document 32-9154522-002, "Oconee MUR Fluence," June 2011

IV.6 AREVA documents 51-9159300-000, "ONS Units, 1, 2, and 3 MUR Power Uprate - RCS Structural Assessment," May 2011

IV.7 OLRP-1001, "Application for Renewed Operating Licenses, Oconee Nuclear Station, Units 1, 2, and 3," Revision 2, July, 1998

IV.8 NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station Units 1, 2 and 3," March, 2000

IV.9 BAW-2241, "Fluence and Uncertainty Methodologies," Revision 1, December, 1999

IV.10 NRC letter from Frank Akstulewicz to J.J. Kelly (B&W Owner's Group), "Acceptance for Referencing of Licensing Topical Report BAW-2241-P, "Fluence and Uncertainty Methodologies" (TAC NO. M98962)," undated

IV.11 USNRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Rev 0, March 2001

IV.12 AREVA document 51-9164100-000, ONS Units 1, 2 & 3: Assessment of RV Beltline Equivalent Margin Analysis at 48 EFPY with MUR Power Uprate, July 2011

IV.13 AREVA document 51-9163792-000, ONS Units 1, 2 & 3: Evaluation of Postulated RV Underclad Cracks at 48EFPY with MUR Power Uprate, July 2011

IV.14 AREVA document 32-9163268-000, "Oconee Units 1, 2 and 3: C_VUSE Calc. Due to MUR Power Uprate 48 EFPY," July 2011

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: All electrical systems at Oconee 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.

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.

AC Distribution

The following AC distribution systems were reviewed:

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

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).

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

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 Oconee is the Keowee Hydro Station.

The Keowee Emergency Power System is designed to provide a reliable emergency onsite power source for the Oconee Nuclear Station. The system consists of the Keowee Hydro Station, a 13.8kV underground cable feeder to Transformer CT4, and a 230kV transmission line to the 230kV switching station at Oconee 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 stepup 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 provide emergency electrical power for the plant Engineered Safeguard Features plus selected balance of plant emergency loads. The MUR uprate will not change the loading of the Keowee Hydro Units. Therefore, Keowee Generator System equipment capacity and capability for plant operations under MUR power uprate conditions are bound by the loading tables, which are supported by the existing analysis of record. As a result, the Keowee Generator 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. Electrical power is available immediately from the battery systems and within 10 minutes from the SSF diesel generator. The MUR 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 bound 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 Oconee Environmental Qualification (EQ) Program is guided by the regulations detailed in 10 CFR 50.49, IE Bulletin 79-01B, Regulatory Guide 1.97, and NUREG-0737 Supplement 1. Duke Energy has reviewed the Oconee EQ program for the MUR Power Uprate and determined that no EQ Program changes are required as a result of the MUR Uprate. In accordance with the ONS design change process, any specific component modifications that may be required to support the MUR Uprate will be evaluated against the EQ Program requirements.

V.1.D grid stability

RESPONSE: The main electrical generators were reviewed at each of the Oconee Units and it was determined that the electrical generators are acceptable for the MUR power uprate. The increase of MWe due to the MUR 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 actual/available MW/MVAR loading before and after the MUR is given below. The electrical generators are therefore acceptable for the MUR power uprate.

	ONS Unit 1		ONS Unit 2		ONS Unit 3	
Turbine nameplate rating: 1037.937 MVA, 0.90 PF						
	MW	MVAR	MW	MVAR	MW	MVAR
Equivalent nameplate rating	934	452	934	452	934	452
Current (typical) values	906.5	475	907.3	475	914.8	460
Post-MUR (typical) values	924.6	460	924.8	460	932.5	455

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

1) Thermal Analysis Study
2) Fault Duty Study
3) Stability Study
4) Reactive Capability Study

Thermal Analysis Study:

Duke Energy power grid Thermal Analysis Study conservatively evaluated future capacity during the warmest period of the year. Estimated loading throughout the system was studied for the effects it has on individual grid elements: basically what elements are operating near full capacity. Power available to be supplied to system loads was evaluated based on each ONS generator producing the expected post-MUR increased output. The results of the grid thermal study show that the increase in loading from the MUR is within the capability of the network.

Fault Duty Study:

The Fault Duty Study is based on the grids maximum design limits. Simulated symmetrical and asymmetrical shorts are assumed at strategic points in the system model and the fault currents are analyzed with respect to equipment current and time-current characteristics.

The existing fault study values bound the increase in loading due to the MUR uprate, and will have no impact on the fault study. The impedance of the machine (i.e. the Oconee generators) has not changed. Consequently, the existing Fault Duty Study remains bounding and was not re-performed.

Stability Study:

Stability studies are relational studies to evaluate generation equipment response characteristics with respect to a stimulating event. Usually stability analyses take the form of evaluating the system with various shorts inserted at strategic points and studying the time and frequency related response of the system, a step response to a system impedance change.

The existing stability analysis bounds the post-MUR generation equipment characteristics such that existing stability analysis remains applicable. The stability study was not re-performed because the generation addition does not provide a material change to the overall stability of the system. Previous studies are still applicable and do not indicate any stability concerns.

Reactive Capability Study:

Reactive capability studies evaluate the capability of the generator and downstream components to generate or carry VARs. With the MUR increase, the Oconee generators will have slightly less VAR capability based on the generator capability curves. The Isolated Phase Bus (IPB) will continue to limit the VAR output of the Oconee generators as it does at present power levels. However, the grid system reactive power capability remains acceptable because adequate reactive support exists in the region.

The results of these four studies show that grid stability is not impacted by the MUR uprate.

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 VI-1 contains a summary of the typical BOP operating parameters before and after the MUR uprate. As can be seen from Table VI-1, the operating parameter changes as a result of the MUR uprate are small.

Table VI-1 - Typical BOP Operating Conditions before and after the MUR uprate

		Winter Operation		Summer Operation	
Description	units	2568 MWt	2610 MWt	2568 MWt	2610 MWt
Circ Water Inlet Temp	deg F	48	48	85	85
SG Outlet Pressure	psia	910	910	910	910
SG Outlet Temperature	deg F	592.3	591.8	593.0	592.5
SG Outlet Steam Flow	10^6 lbm/hr	10.80	11.02	10.80	11.01
Final FW Temp	deg F	455.7	457.5	455.7	457.5
Final FW Pressure	psia	1011.9	1003.6	1008.7	1000.4
Gross Generator Output	MWe	914.1	928.9	882.8	896.9

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:

Main Steam:

The Oconee Main Steam (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 Oconee.

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 uprate to the current 2568 MWt conditions demonstrates that the Main Steam System has sufficient design and operational margin to accommodate the MUR. The MUR power uprate conditions remain bounded by the design basis of record.

Condensate and Vacuum:

The condensate/vacuum systems includes the condenser, hotwell pumps, condensate booster pumps, low pressure feedwater heaters, upper surge tanks and condensate coolers. The only safety function of these systems is for the condensate system to provide a source of water to the Emergency Feedwater System.

A comparison between operating conditions for the 2610 MWt MUR uprate to the current 2568 MWt conditions demonstrates that the Condensate and Vacuum Systems have sufficient design and operational margin to accommodate the MUR uprate. The MUR power uprate conditions remain bounded by design as described in the Oconee UFSAR.

Main Feedwater:

The main feedwater (FDW) system includes the high pressure heaters, the feedwater pumps, and feedwater control valves. The FDW system has one safety function to provide feedwater isolation as required by Technical Specification 3.7.3.

A comparison between operating conditions for the 2610 MWt MUR uprate to the current 2568 MWt conditions demonstrates that the FDW System (with the exception noted below) has sufficient design and operational margin to accommodate the MUR uprate. The MUR power uprate conditions remain bounded by design as described in the Oconee UFSAR.

The FDW Pump Turbines require modifications prior to operating at MUR conditions. The current bucket configuration in the Oconee Main Feedwater Pump Turbines (FWPTs) is subject to cyclical loading that induces stresses above the recommended values. The MUR operating conditions will increase turbine running speed which results in an increase in the stress on the buckets. This condition will require a modification to the FWPTs in order to support the uprate. These equipment modifications will be completed prior to operating at the uprated power level. (ONS will not operate at MUR conditions until at least the 6th stage blades are replaced on the FWPTs. Unit 1 blade replacements are complete. Unit 2 and Unit 3 are being evaluated. Oconee Units 2 and 3 will not implement the MUR power uprate until either the 6^{th} stage blade replacements, or other modification, is completed). This commitment is included in Attachment 1 of this LAR.

Emergency Feedwater:

The major components in the EFW System of each unit include two motor-driven EFW pumps, one turbine-driven EFW pump, two automatic EFW flow control valves, one ASW pump, and various isolation valves. The Auxiliary Service Water portion of the EFW System is shared among all three units.

The EFW System has the following safety functions:

Provide an assured source of feedwater to the SGs to remove decay heat until the LPI System may be operated or the MFW System is restored. This function is credited for LOOP, Locked Rotor Shaft Seizure (LRSS), MSLB, REA, SBLOCA, and SGTR events.

Provide isolation of EFW flow following a Main Steam Line Break or SG Tube Rupture or to prevent dilution of the Reactor Building Sump. This function is credited for MSLB and SGTR events.

Provide shell cooling of an isolated SG during plant cooldown when the MFW system is unavailable. This function is credited for MSLB, REA, SBLOCA, and SGTR events.

Prevent EFW pump runout and SG tube flow-induced vibration. This function is credited for LOOP, LRSS, MSLB, REA, SB LOCA, and SGTR events.

Supply raw water to SGs for decay heat removal. This function is credited for the tornado event. Specifically, the ASW pump is credited for the tornado event.

Provide EFW to SGs in case of low SG level to minimize dryout.

Provide backup EFW to other Oconee units.

Maintain containment integrity.

Supply backup cooling water to HPI pump motor coolers (ASW).

Provide backup power to the HPI pump motors. (This is an electrical function of the ASW switchgear.)

The current design and capabilities of the Emergency Feedwater / Auxiliary Service Water systems and components remain bounding for the MUR conditions. New MUR conditions do not impose any necessary changes to system design flow rates, volumes, temperatures or pressures.

Auxiliary Steam:

The Auxiliary Steam (AS) System supplies startup steam as necessary when the MS System is not available. Startup steam can be cross -connected between the three Oconee units to allow an operating unit's MS to supply startup steam. A shared auxiliary boiler is provided in the event other sources are not available.

The only safety related function of the AS system is to be available to the Emergency Feedwater Pump Turbine (EFWPT) under normal operating conditions anytime the RCS temperature exceeds 250 F.

A comparison of operating conditions for the 2610 MWt MUR to the current 2568 MWt conditions demonstrates that the Auxiliary Steam System has sufficient design and operational margin to accommodate the MUR.

VI.1.B containment systems

RESPONSE: The containment systems at ONS include:

The building spray system:

This Reactor Building Spray (BS) System 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 (RB) 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 MS 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.

Penetrations and hatches:

The safety function of the penetrations and hatches is to maintain containment integrity under accident conditions. As indicated in Sections II and III 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 uprate.

Coatings in Containment:

The proposed MUR at ONS does not have any impact to the programmatic aspects of the Coating 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 is bounded by current analysis of record and no changes are required.

Ventilation systems:

The reactor building ventilation system consists of the reactor building ventilation (air) system, the penetration room ventilation system, and the spent fuel pool ventilation system (a filtration system as well as a cooling system). Even though there are separate ventilation systems, the LPSW system provides water to all cooling coils serving the reactor and SFP areas.

A comparison of the operating conditions for the 2610 MWt MUR uprate to the current 2568 MWt conditions demonstrates that the reactor building ventilation systems have sufficient design and operational margin to accommodate the MUR. The systems remain bounded by the existing analyses of record.

Reactor building purge system:

The Reactor Building Purge (PRP) system provides the Reactor Building with fresh air during outages to reduce airborne contaminant levels inside the Reactor Building. The PRP 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 PRP system containment penetrations are required to be sealed closed thereby prohibiting operation of the system.

The only safety function of the PRP system is containment isolation. The PRP System will continue to provide containment isolation after the MUR power increase.

The Reactor Building Purge System will continue to operate in order 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. The Reactor Building Purge 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.

Reactor building cooling system:

The Reactor Building Cooling (RBC) System provides cooling to limit the temperature and pressure of containment following a design bases event (DBE).

The RBC performs the safety-related function of maintaining reactor building temperature and pressure within the environmental qualification limits following design basis accidents.

The RBC System will continue to perform its safety related functions of maintaining containment temperature and pressure within environmental qualification limits 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.C safety-related cooling water systems

RESPONSE:

Component Cooling System:

The Component Cooling (CC) System provides closed loop cooling water to various heat exchangers located inside the Reactor Building. The heat that is transferred to the Component Cooling 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 are 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.

Condenser circulating water:

The Condenser Circulating Water (CCW) System 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. Insure suction remains provided to LPSW Pumps during loss of power to all CCW Pumps from ECCW siphons.

- provide a suction source for Station ASW Pump via water in the Unit 2 inlet piping. The CCW System shall be capable of transferring water from all three units' CCW inlet and discharge piping to support 37 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. Insure 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 2 hours during a fire.

The MUR uprate will have no impact on 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 conditions.

The MUR uprate will have no impact on the ASW System suction source. Suction to the ASW 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 conditions. The water available to ASW in the CCW piping will continue to last approximately 37 days because the volume available in the piping is unchanged due to MUR and the volume required is based on decay heat loads assuming 102% of 2568 MWt.

The MUR 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 conditions. Also the CCW System configuration is not impacted by the MUR uprate; therefore, the CCW System can still insure a flow path to the SSF submersible pumps is maintained.

The MUR 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 conditions. Also the CCW System configuration is not impacted by the MUR 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 and the current 2568 MWt conditions demonstrates that the Condenser Circulating Water System has sufficient design 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 system remains bounded by the existing analysis of record.

High pressure service water:

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

The HPSW system performs the following safety functions and regulatory requirements.

- prevent air in-leakage from air binding the low pressure service water pumps if the elevated water storage tank is depleted. This is a safety function.

- 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. This is a regulatory requirement.

The safety 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.

Low pressure service water:

The LPSW System 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 coolers, the reactor building cooling units, and the secondary service water headers.

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

Standby Shutdown Facility Auxiliary Service Water (SSF ASW)

The SSF ASW 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 following various events.

The SSF ASW system has no functions related to the design basis events described in Chapter 15 of the UFSAR. For other events, 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 Auxiliary Service Water System and components remain bounding for MUR conditions. New MUR conditions do not impose any necessary changes to system design flow rates, volumes, temperatures or pressures.

Protected Service Water

Duke Energy is currently installing a Protected Service Water (PSW) system that 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 a Safety-related system designed to be manually aligned to the required unit in the event all other means of maintaining steam generator feedwater sources are unavailable to support RCS cooldown. The requirement for this system is the result of the reconstituted HELB licensing basis. PSW system operating parameters are base on the revised HELB analysis performed at 2619 MWt or 102% of the original core power rating of 2568 MWt.

The PSW system will have the following safety functions (Note that when PSW is installed, this may change the safety functions of the systems that currently perform these functions.):

- Assure natural circulation and core cooling by providing secondary side cooling water from Lake Keowee.
- Transfer decay heat from the fuel to an ultimate heat sink.
- Maintain the reactor 1% shutdown with the most reactive rod stuck fully withdrawn, after all normal sources of RCS makeup have become unavailable, by providing makeup via the High Pressure Injection (HPI) system which supplies makeup of a sufficient boron concentration from the Borated Water Storage Tanks (BWST).
- Be able to control the above functions from the Main Control Rooms.

The PSW System will perform its safety related functions after the MUR power uprate. The analyses of record were performed at 2619 MWt (102% of 2568 MWt); therefore, the analysis of record bounds the flow, temperature and pressure required by the PSW System and its components following the MUR power uprate. There is no impact to the design basis or operation of this system due to the MUR.

Recirculating Cooling Water System (a Non-safety related cooling water system):

The Recirculating Cooling Water (RCW) System 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 is capable of supporting the MUR design conditions. The RCW System design flow rate capacity provides adequate design flow to the components within the system at both current and MUR conditions. The RCW system cools one Isolated Phase Bus (IPB) Air Cooler for each unit. These coolers cool the air that circulates across the Isolated Phase Buses. The IPB ventilation systems for Oconee Units 1, 2, & 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 must either provide supplemental cooling or limit the maximum thermal power. During these conditions, the full potential of the MUR uprate may not be realized.

VI.1.D spent fuel pool storage and cooling systems

RESPONSE:

The Refueling System (RFS)

The Refueling System (RFS) 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 Oconee. 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 will be maintained in the storage racks in a subcritical condition. There is no impact to this system due to the MUR.

The spent fuel cooling (SF) system

The spent fuel cooling (SF) system is composed of six pumps, nine 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 Oconee licenses for Units 1, 2 & 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 irradiated to a 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.

VI.1.E radioactive waste systems

RESPONSE: A brief review was made of the radioactive waste 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.

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.

VI.1.F Engineered safety features (ESF) heating, ventilation, and air conditioning systems

RESPONSE: The chilled water - vital loads (WC) system and the control room ventilation system (CRVS):

The WC system provides chilled water to ensure the heat loads, considered vital loads, are removed from the areas served by the control room ventilation system (CRVS) when air is circulated via the respective cooling coils of the associated air handling units. There are two separate CRVS systems (one for Units 1 and 2 and a separate CRVS for Unit 3). 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 system has a safety function to provide cooling and filtration to the operators and equipment in the control room, cable room and equipment room.

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

VI.1.G Fire Protection Systems

RESPONSE: The MUR power uprate was reviewed against the Oconee Fire Protection Program, including the current Appendix R Safe Shutdown Analysis and the design documents that support the transition to NFPA 805. The MUR 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. No changes were made to the plant configuration or combustible loading as a result of implementing the MUR uprate that affect the ONS fire protection program. Additional building heat up will be minimal such that currently credited Appendix R manual actions and future NFPA 805 recovery actions will not be prevented from being accomplished within their required time.

The impact of the MUR uprate on the Appendix R / NFPA 805 analyses of record are discussed in Section II of this enclosure.

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 proposed MUR power uprate will be implemented under the administrative controls of Oconee Nuclear Station design change process. 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 if required. An evaluation was performed of the Operator Actions and no impacts were identified.

Time Critical Operator Actions (TCOA) 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 design basis events, regulatory commitments, and events with high Probabilistic Risk Assessment values. The TCOA were evaluated individually in system evaluations. In addition the TCOA were evaluated against the Oconee licensing analyses presented in Sections II and III of this enclosure to ensure they remain bounded. All of 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 uprate will be implemented under the administrative controls of Oconee Nuclear Station design change process. The design change process ensures any impacted emergency and abnormal operating procedures are revised prior to the implementation of the MUR.

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 [e.g., software modification that redefines the new 100% (2610 MWt RTP)]. Oconee Nuclear Station follows the established engineering procedures to ensure the necessary minor modifications are installed prior to implementing the proposed MUR.

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.

VII.2.C the 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 Oconee Nuclear Station design change process. As part of this process, any necessary changes to the simulator are identified and implemented during the design change review process.

VII.2.D the operator training program

RESPONSE: The Operator training program will be modified to reflect the MUR. Operator training on the plant changes required to support the MUR will be completed prior to MUR implementation.

Training on operation and maintenance of the Caldon LEFM CheckPlus System, will be developed and completed prior to implementation of the MUR 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: All changes/modifications to the simulator and the associated manuals and instructional materials will be implemented in accordance with the Oconee engineering change process to capture all plant changes as a result of the MUR uprate. Duke Energy will complete all modifications related to the MUR and complete the training of operators, prior to implementation of the 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: ONS Operating Procedures (OPs) have been reviewed. It is Duke Energy policy never to intentionally operate above rated thermal power. While Minor fluctuations of instantaneous indicated thermal power, the five minute average, and the 15 minute average, are expected, operator guidance is to reduce power whenever power is above rated thermal power.

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:

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 Keowee dam.

The NPDES Permit identifies both chemical and thermal discharge limits for the plant. The MUR uprate includes no plans to 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 MUR. 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.

Based on the previous NPDES permit, the current NPDES permit reports the following historic thermal discharge conditions:

Average and Maximum Daily Flow Values for Outfall 001: A long-term average of 2519.9 million gallons per day (mgd) and a Maximum Daily and Maximum 30-Day flow of 3058.6 mgd, representing a combined flow for all three Units at Oconee.

Thermal discharge values in the NPDES Permit Fact Sheet are defined as follows:

Discharge Temperature: Daily Maximum: 100°F (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)

Daily Maximum: 103°F (when 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).

Delta T Daily Maximum: 22°F (5.55°C) [this limit does not apply when the intake temperature is less than 68°F (20.0°C); Delta T can be controlled by the number of CCW pumps in operation.]

The Oconee plants have also received a variance under Section 316(a) of the Clean Water Act (CWA), regarding thermal effluent cooling water. CWA Section 316(a) allows for a thermal discharge variance based on a demonstration that less stringent thermal effluent limitations would still protect aquatic life. Results of studies of relevant discharge water and lake temperatures for the years 1999 through 2005, including evaluation of the potential impacts on local aquatic populations, are considered in the determination of NPDES thermal discharge limits.

Daily average intake and discharge temperatures reported for NPDES Permit requirements during the most recent ten years of plant operation show a maximum daily average high of 83.86°F for that period. High intake water temperatures (above 83°F) occur only intermittently and persist for only a few days each in the ten year record.

Measured discharge temperatures show a maximum daily average high of 99.32°F for the period of ten years. High discharge water temperatures (above 99°F) occur only intermittently and at most for only a week or two at a time in the ten year record.

The plant cooling water systems do not approach the NPDES Delta-T requirement for intake water above 68°F. Delta-T values above 22°F occur only occasionally, in cold or cooler months and generally when intake water is in the mid-50°F range.

An assessment of the MUR 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 0.2 °F. This 0.2 °F increase will not cause NPDES permit limits to be exceeded. Administrative controls are in place to temporarily reduce power levels should the discharge temperatures encroach on NPDES permit thermal limits.

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. Waste Liquid and Gaseous data for licensed releases performed to date, and assumed to be representative for future releases, indicate that doses are a small fraction of allowable annual limits. The data provides verification that the MUR will not cause doses from waste liquid and gaseous waste releases to exceed allowable limits.

The presence of tritium in plant groundwater was also considered as a potential issue for the MUR. While a specific cause for this issue is not yet determined for Oconee, U.S. NRC Regulatory Guide 4.21 presents a comprehensive scope of potential causal factors in considering factors to mitigate potential for this condition. None of these factors are relevant to plant power generation levels. Thus, that environmental issue is not considered relevant for the MUR.

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

RESPONSE: A significant increase in individual and cumulative occupational radiation exposure is unlikely because of the low percentage power increase involved. Individual worker exposures will be maintained within acceptable limits by the station Radiation Protection and As Low As Reasonable Achievable (ALARA) Programs. Thus, no radiological environmental impact is anticipated. Dose evaluations for accident scenarios reported in Section 15 of the Oconee UFSAR already take into account, as applicable, an operating level of 102% of the baseline plant power rating as were provided by Appendix K to 10CFR50.

As explained below, current ALARA requirements and Radiological Protection controls remain in effect. As a result, dose is not anticipated to be impacted by the MUR.

Individual worker exposures will be maintained within acceptable limits by the station Radiation Protection and ALARA Programs. Baseline worker dose analyses for the 102% power level are bounded by the existing analyses of record. Thus, no impact on radiological dose is anticipated.

References for Section VII:

VII.1. DHEC, 2010, NPDES Permit SC0000515, Duke Energy Carolinas LLC, Oconee Nuclear Station, Oconee County, Issued March 30, 2010

VII.2 Regulatory Guide 4.21, Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning, June 2008

VII.3 Title 10 Code of Federal Regulation, Part 50 (10 CFR 50), Appendix K, "ECCS Evaluation Models"

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 3 of Enclosure 1. Revised Technical Specifications are attached, a marked up copy in Attachment 2 and a retyped copy in Attachment 4. Likewise, marked up Technical Specification bases are provided in Attachment 3 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 uprate Technical Specification changes, or are affected by the MUR uprate Technical Specification changes, are discussed below.

The heat balance uncertainty calculation, Duke Energy calculation OSC-3737, 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, and a copy of the calculation is included in Attachment 6 to this LAR.

The RPS overpower setpoint calculation has been revised based on the increased rated thermal power. The results of this change are reflected in the revised Technical Specification setpoints in TS Table 3.3.1-1. No changes in methodology were used for this revision. The revised trip setpoints are bounded by the values used in the accident analyses discussed in Sections II and III of this Enclosure.

An RPS overpower setpoint allowable value (79.3% RTP) has been added to the Nuclear Overpower high setpoint allowable values for 3 reactor coolant pumps (RCPs) operating. The basis for the new 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.

The Flux-Flow-Imbalance setpoints were also reviewed based on the increased power level, and it was determined that the flux-flow-imbalance envelope could remain unchanged.

The calculation of the arming setpoint for ATWS 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 the Technical Specification Bases changes in Section 3 of Enclosure 1.

References for Section VIII:

VIII.1 Duke Energy Calculation OSC-3737, Rev. 9, Secondary Power Uncertainty Analysis, 02 Feb 2011

ATTACHMENT 1

REGULATORY COMMITMENTS

The following commitment table identifies those actions committed 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.

	Commitment	Completion Date
1	Any revisions to setpoint calculations or calibration procedures necessary to reflect the increased rated thermal power will be implemented. All maintenance procedures for the new equipment added for the MUR uprate will be implemented.	Prior to implementation of the MUR power uprate.
2	Duke Energy will not operate at the uprated power level until required Feedwater pump steam turbine modifications are completed on each Unit.	Prior to operating at the uprated power level.
3	Duke Energy will complete all training of operators on the changes related to the MUR power uprate.	Prior to implementation of the MUR power uprate.
4	Duke Energy will position the ONS Unit 2 and Unit 3 leading edge flowmeters in compliance with Cameron requirements to ensure there is no impact on the existing venturis. Location of the LEFMs will be determined during development of the Engineering Change Package for Units 2 and 3.	Prior to implementation of the MUR power uprate.
5	Duke Energy will develop maintenance procedures for the Cameron equipment, and train maintenance personnel on those procedures, prior to implementation of the MUR.	Prior to implementation of the MUR power uprate.
6	Duke Energy will complete the double main steam line break mitigated by the SSF and make it available for NRC review by October 31, 2011.	October 31, 2011.
7	Duke Energy will implement a Selected Licensee Commitment to provide functionality requirements for the leading edge flow meters with appropriate Required Actions and Completion Times when the LEFM is not functional.	Prior to implementation of the MUR power uprate.
8	Duke Energy will implement a Selected Licensee Commitment to provide a setpoint methodology for the RPS functions with Allowable Value affected consistent with TSTF 493, Option A	Prior to implementation of the MUR power uprate.

ATTACHMENT 2

TECHNICAL SPECIFICATION MARKUPS

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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. ~~370~~, 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

XXX

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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.

YYY

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.~~372~~, 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

YYY

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

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

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~371~~ 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

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~~ 2610 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 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.

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 – 4 reactor coolant pumps running	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.2(d)(e) SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 105.5% RTP(f)
~~b.~~ c. 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 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 618°F
b. High Setpoint – 3 reactor coolant pumps running	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.2(d)(e) SR 3.3.1.4 SR 3.3.1.5	≤ 79.3% RTP(f)
4. RCS Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.4 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 3.3.1.4 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
6. Reactor Building High Pressure	1,2,3(c)	C	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.5 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 3.3.1.4 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 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.4 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 3.3.1.4 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 3.3.1.4 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)

(c)

(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 that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the limiting Trip Setpoint or a value that is more conservative than the Limiting Trip Setpoint; otherwise the channel shall be declared inoperable. The limiting Trip Setpoint and the methodology used to determine the limiting Trip Setpoint, the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Selected Licensee Commitments.

(f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in the Selected Licensee Commitments.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

1.

73.8

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating;
b. Three RCPs operating and:
THERMAL POWER restricted to ≤~~75~~% RTP.

2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.b (Nuclear Overpower – High Setpoint for 3 RCP Operation), Allowable Value of Table 3.3.1-1 is reset for 3 RCPs operating; and
3. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 8 (Nuclear Overpower Flux/Flow/Imbalance), Allowable Value specified in the COLR is reset for 3 RCPs operating.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ~~A.~~ Required Action and associated Completion Time of Condition A not met.	B. ~~A.~~1 Be in MODE 3.	12 hours

OR

Requirements of LCO not met for reasons other than Condition A.

A. Requirements of LCO 3.4.4.b.2 not met	A.1 Reset the RPS to satisfy the requirements of LCO 3.4.4.b.2.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

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ATTACHMENT 3

TECHNICAL SPECIFICATION BASES MARKUPS

BACKGROUND RPS Overview (continued)

These arrangements and the relationship of instrumentation channels to trip Functions are discussed next to assist in understanding the overall effect of instrumentation channel failure.

b. Nuclear Overpower - High Setpoint with 3 RCPs

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions:

with 4 RCPs

1. Nuclear Overpower

   a. Nuclear Overpower – High Setpoint;

   c. ~~b.~~ Nuclear Overpower - Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower Flux/Flow Imbalance;

9. Main Turbine Trip (Hydraulic Fluid Pressure); and

10. Loss of Main Feedwater (LOMFW) Pump Turbines (Hydraulic Oil Pressure).

The power range instrumentation has four linear level channels, one for each core quadrant. Each channel feeds one RPS protective channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE for the associated core quadrant.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

&b

~~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.

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 units with the MUR power uprate, 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 Measurement Uncertainty Recapture (MUR) power uprate complete, the nuclear overpower setpoint is 105.5% of 2568 MWt, or 2709 MWt. For units with the MUR power uprate, the nuclear overpower setpoint is 105.5% of 2610 MWt, or 2754 MWt.

For Unit(s) with the Measurement Uncertainty Recapture (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

&b

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~a.~~ Nuclear Overpower – High Setpoint (continued)

rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

A Nuclear Overpower – High Setpoint value is also provided for 3 RCP operation following the MUR uprate. The purpose for the 3 RCP trip is to provide protection for power excursion events initiated from 3 RCP operation, most notably the small steam line break accident.

~~b.~~ c

Nuclear Overpower – Low Setpoint

When initiating shutdown bypass, the Nuclear Overpower – Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

Amendment Nos. XXX, YYY, ZZZ

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

10. Loss of Main Feedwater Pump Turbines (Hydraulic Oil Pressure (continued)

For the feedwater pump turbine hydraulic oil pressure, the Allowable Value of 75 psig is selected to provide a trip whenever feedwater pump turbine hydraulic oil pressure drops below the normal operating range. This trip is bypassed at power levels < 2% RTP for unit startup. The Loss of Main Feedwater Pump Turbines (Hydraulic Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass RCS High Pressure is provided to allow for withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. The shutdown bypass provides trip protection during deboration and RCS heatup by allowing the operator to at least partially withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear Overpower – High Power trip setpoint, and again withdraw the safety group rods before proceeding with startup.

Accidents analyzed in the UFSAR, Chapter 15 (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the UFSAR.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1720 psig and the Nuclear Overpower – Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

1a. Nuclear Overpower – High Setpoint for 4 RCP operation;

1b. Nuclear Overpower - High Setpoint for 3 RCP

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

General Discussion

The RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 7). In MODES 1 and 2, the following trips shall be OPERABLE because the reactor can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during anticipated transients and to assist the ESPS in providing acceptable consequences during accidents.

1a. Nuclear Overpower – High Setpoint;

for 4 RCP operation

1b. Nuclear Overpower - High Setpoint for 3 RCP

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

Functions 1a, 1.b, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below 1720 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower – Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

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.

73.8

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. With three pumps in operation the reactor power level is restricted to ~~75~~% RTP 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.

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 of 112% of 2568 MWt, = 2876 MWt. This is the design overpower condition for four pump operation. The 105.5% RTP value is the setpoint of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors.~~ 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.

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 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 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).
~~RCS Loops – MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 2)~~

as must the Nuclear Overpower – High Setpoint – 3 reactor coolant pumps running

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.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, the reactor is critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";

LCO 3.4.6, "RCS Loops – MODE 4";

LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";

LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";

LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and

LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

reset the Nuclear Overpower - High Setpoint and the Nuclear Overpower Flux/Flow/Imbalance Setpoints to satisfy the requirement of LC0 3.4.4.b.2

3.4.4.b.2

A.1

If the requirements of the LCO are not met, the Required Action is to ~~reduce power and bring the unit to MODE 3~~. This ~~lowers power level and thus reduces the core heat removal needs and~~ minimizes the possibility of violating DNB limits.

6

The Completion Time of ~~12~~ hours is reasonable, based on operating experience~~, to reach MODE 3 from full power conditions~~ in an orderly manner and without challenging safety systems.

reset the RPS setpoints

B.1

If the required time and associated completion time of Condition A is not met or the requirements for the LCO are not met, the Required Action is to reduce power and bring the unit to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requ[...] loops in oper[...] status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

Amendment Nos. XXX, YYY, ZZZ

ATTACHMENT 4

RETYPED TECHNICAL SPECIFICATIONS

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

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 2610 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 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.

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– 4 reactor coolant pumps running	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.2[d][e] SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 105.5% RTP[f]
b. High Setpoint – 3 reactor coolant pumps running	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.2[d][e] SR 3.3.1.4 SR 3.3.1.5	≤ 79.3% RTP[f]
c. 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 3.3.1.4 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 3.3.1.4 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 3.3.1.4 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 3.3.1.4 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 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 4 psig

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
7. Reactor Coolant Pump to Power	1,2[(a)]	C	SR 3.3.1.1 SR 3.3.1.4 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 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.4 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 3.3.1.4 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 3.3.1.4 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 that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the limiting Trip Setpoint or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The limiting Trip Setpoint and the methodology used to determine the limiting Trip Setpoint, the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Selected Licensee Commitments.

(f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in the Selected Licensee Commitments.

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 restricted to ≤73.8% RTP.
   2. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 1.b (Nuclear Overpower – High Setpoint for 3 RCP Operation), Allowable Value of Table 3.3.1-1 is reset for 3 RCPs operating; and
   3. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," Function 8 (Nuclear Overpower Flux/Flow/Imbalance), Allowable Value specified in the COLR 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.2 not met	A.1 Reset the RPS to satisfy the requirements of LCO 3.4.4.b.2	6 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

ATTACHMENT 5

RETYPED TECHNICAL SPECIFICATION BASES

BACKGROUND

RPS Overview (continued)

These arrangements and the relationship of instrumentation channels to trip Functions are discussed next to assist in understanding the overall effect of instrumentation channel failure.

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions:

1. Nuclear Overpower
    a. Nuclear Overpower – High Setpoint with 4 RCPs;
    b. Nuclear Overpower – High Setpoint with 3 RCPs;
    c. Nuclear Overpower – Low Setpoint;

7. Reactor Coolant Pump to Power;
8. Nuclear Overpower Flux/Flow Imbalance;
9. Main Turbine Trip (Hydraulic Fluid Pressure); and
10. Loss of Main Feedwater (LOMFW) Pump Turbines (Hydraulic Oil Pressure).

The power range instrumentation has four linear level channels, one for each core quadrant. Each channel feeds one RPS protective channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE for the associated core quadrant.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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. <u>Nuclear Overpower</u>

a & b. <u>Nuclear Overpower – High Setpoint</u>

The Nuclear Overpower – High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core neutron leakage flux.

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 units with the MUR power uprate, 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 Measurement Uncertainty Recapture (MUR) power uprate complete, the nuclear overpower setpoint is 105.5% of 2568 MWt, or 2709 MWt. For units with the MUR power uprate, the nuclear overpower setpoint is 105.5% of 2610 MWt, or 2754 MWt.

For Unit(s) with the Measurement Uncertainty Recapture (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.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a & b. Nuclear Overpower – High Setpoint (continued)

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.

A Nuclear Overpower – High Setpoint value is also provided for 3 RCP operation following the MUR uprate. The purpose for the 3 RCP trip is to provide protection for power excursion events initiated from 3 RCP operation, most notably the small steam line break accident.

c. Nuclear Overpower – Low Setpoint

When initiating shutdown bypass, the Nuclear Overpower – Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear Overpower – High Power trip setpoint, and again withdraw the safety group rods before proceeding with startup.

Accidents analyzed in the UFSAR, Chapter 15 (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the UFSAR.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1720 psig and the Nuclear Overpower – Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

1a. Nuclear Overpower – High Setpoint for 4 RCP operation

1b. Nuclear Overpower – High Setpoint for 3 RCP operation

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

General Discussion

The RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 7). In MODES 1 and 2, the following trips shall be OPERABLE because the reactor can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during anticipated transients and to assist the ESPS in providing acceptable consequences for 4 RCP operation during accidents.

1a. Nuclear Overpower – High Setpoint for 4 RCP operation;

1b. Nuclear Overpower – High Setpoint for 3 RCP operation

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

Functions 1a, 1b, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below 1720 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower – Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

The Main Turbine Trip (Hydraulic Fluid Pressure) Function is required to be OPERABLE in MODE 1 at ≥ 30% RTP. The Loss of Main Feedwater Pump Turbines (Hydraulic Oil Pressure) Function is required to be OPERABLE in MODE 1 and in MODE 2 at ≥ 2% RTP. For operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

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. With three pumps in operation the reactor power level is restricted to 73.8% RTP 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.

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 73.8% 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 running.

BASES (continued)

APPLICABILITY

In MODES 1 and 2, the reactor is critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";

LCO 3.4.6, "RCS Loops – MODE 4";

LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";

LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";

LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation – High Water Level" (MODE 6); and

LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO 3.4.4.b.2 are not met, the Required Action is to reset the Nuclear Overpower - High Setpoint and the Nuclear Overpower Flux/Flow/Imbalance Setpoints to satisfy the requirement of LC0 3.4.4.b.2. This minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reset the RPS setpoints in an orderly manner and without challenging safety systems.

BASES (continued)

B.1

If the required time and associated completion time of Condition A is not met or the requirements for the LCO are not met, the Required Action is to reduce power and bring the unit to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. UFSAR, Chapter 15.

2. 10 CFR 50.36.

Attachment 6-1

Cameron affidavit requesting proprietary treatment of ER-813, Rev 1, ER-824 Rev 1, ER-825 Rev 1, and ER-855

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
Tel 724-273-9300
Fax 724-273-9301
www.c-a-m.com

CAMERON

June 28, 2011

Attention: Terry Bradley
Duke Energy
526 South Church St.
Charlotte, NC 28202

Phone: (704) 382-5997

Subject: Application for withholding proprietary information from public disclosure.

Dear Terry,

Per your request please find enclosed your application for withholding proprietary information from public disclosure for McGuire Units 1 & 2 CAW 11-04 and Oconee Units 1, 2 & 3 CAW 11-05.

If you have any questions or require additional information, please contact me at 724-273-9300 or Garrett.McLean@c-a-m.com.

Sincerely,

Garrett McLean
Inside Sales

Enclosure

Measurement Systems

Caldon® Ultrasonics Technology Center
1000 McClaren Woods Drive
Coraopolis, PA 15108
Tel 724-273-9300
Fax 724-273-9301
www.c-a-m.com

CAMERON

June 28, 2011
CAW 11-05

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-813 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus System"
2. Caldon® Ultrasonics Engineering Report ER-824 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus System"
3. Caldon® Ultrasonics Engineering Report ER-825 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus System"
4. Caldon® Ultrasonics Engineering Report ER-855 Rev. 0 "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2 and 3"

Gentlemen:

This application for withholding is submitted by Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains trade secrets and/or commercial information proprietary to Cameron and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 11-05 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 11-05 and should be addressed to the undersigned.

Very truly yours,

Ernest Hauser
Director of Sales

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

June 28, 2011
CAW 11-05

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Ernest Hauser, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Cameron International Corporation, a Delaware Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Ultrasonics Technology Center, and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Ernest Hauser
Director of Sales

Sworn to and subscribed before me

this 28th day of

June, 2011

Joann B. Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2011
Member, Pennsylvania Association of Notaries

June 28, 2011
CAW 11-05

1. I am the Director of Sales of Caldon Ultrasonics Technology Center, and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Cameron.

2. I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Cameron application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information. The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

4. Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by 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.

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:

- Caldon® Ultrasonics Engineering Report ER-813 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-824 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-825 Rev. 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus System"
- Caldon® Ultrasonics Engineering Report ER-855 Rev. 0 "Meter Factor Calculation and Accuracy Assessment for the LEFM CheckPlus Meters at Oconee Units 1, 2 and 3"

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 Unit 1, Oconee Unit 2 and Oconee Unit 3 for MUR UPRATES.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Cameron because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Cameron effort and the expenditure of a considerable sum of money.

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Attachment 6-6

Duke Energy affidavit requesting proprietary treatment of OSC-3737

AFFIDAVIT OF D. A. BAXTER

1. I am Vice President of Duke Energy Carolinas, LLC (Duke Energy) and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The Duke Energy proprietary information sought to be withheld in the submittal is Duke Energy Calculation OSC-3737, Revision 9, "Secondary Power Uncertainty Analysis," 02 Feb 2011"

   (ii) The information sought to be withheld from public disclosure is owned by Duke Energy and has been held in confidence by Duke Energy and its consultants.

   (iii) The information is of a type that would customarily be held in confidence by Duke Energy. The information consists of analysis methodology details, analysis results, and supporting data that provides a competitive advantage to Duke Energy.

   (iv) The information is being transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

   (v) The information sought to be protected is not available in public to the best of our knowledge and belief.

(Continued)

D. A. Baxter

This information enables Duke Energy to:

(a) Support license amendment request no. 2011-02 for its Oconee Nuclear Station (Oconee) reactors.

(b) Perform nuclear design calculations on Oconee reactor cores.

(c) Perform transient and accident analysis calculations for Oconee.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.

(a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke Energy can sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke Energy.

5. Public disclosure of this information is likely to cause harm to Duke Energy because it would provide information to vendors that could be used to increase costs charged to Duke Energy and it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

(Continued)

D. A. Baxter

David A. Baxter affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

D. A. Baxter

Subscribed and sworn to me: August 25, 2011
Date

Debra Reese Debra Reese
Notary Public

My Commission Expires: September 15, 2015

SEAL

DEBRA REESE NOTARY PUBLIC MECKLENBURG COUNTY, NC