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Enclosed attachments 5 through 8 contain proprietary and non-proprietary versions of engineering reports. Under 10 CFR 2.390, withhold from public disclosure the proprietary versions only. Upon removal of the proprietary reports, this letter is uncontrolled.

October 30, 2019

10 CFR 50.90

Docket Nos.: 50-348
50-364

NL-19-0795

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

Joseph M. Farley Nuclear Plant Units 1 and 2
Submittal of License Amendment Request for
Measurement Uncertainty Recapture Power Uprate

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Unit 1, Renewed Facility Operating License (NPF-2), and Unit 2, Renewed Facility Operating License (NPF-8), by incorporating the attached proposed changes into the Unit 1 and Unit 2 Technical Specifications (TS) to allow for a measurement uncertainty recovery power uprate (MUR-PU). This MUR-PU license amendment request (LAR) would increase FNP's authorized core power from 2775 megawatts thermal (MWt) to 2821 MWt.

The proposed changes contained herein would revise Operating License (OL) 2.C(1) "Maximum Power Level" and TS Sections 1.1 Definition "Rated Thermal Power (RTP)," 2.1.1 "Reactor Core Safety Limits," 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits," and 5.6.6 "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)."

Note that WCAP-17642-P-A, Rev. 1, "Westinghouse Performance Analysis and Design Model (PAD5)," has been used for this MUR-PU LAR. The use of the PAD5 code and methodology for the not-LOCA analyses as described in this LAR complies with the statements contained in the September 28, 2017 Nuclear Regulatory Commission (NRC) Final Safety Evaluation (SE) for PAD5, including the specific limitations and conditions in Section 4 of the SE.

In addition, WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET," has been utilized to predict the fluence at both the beltline and extended beltline region. However, Limitation and Condition 1 of the NRC safety evaluation approving the WCAP

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requires additional justification when the methodology is applied outside the beltline region. As such, justification for the plant specific application has been provided herein.

The Enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing OL and TS pages marked to show the proposed changes. Attachment 2 provides revised (clean) OL and TS pages. Attachment 3 provides technical information to address the application of WCAP-18124-NP-A in a non-beltline region. Attachment 4 provides the responses to RIS-2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications." Attachments 5 and 6 provide a technical evaluation for the bounding uncertainty analysis for thermal power determination using the leading edge flow meter (LEFM) system for Farley Units 1 and 2, respectively. Attachments 7 and 8 provide meter factor calculations and accuracy assessments for Farley Units 1 and 2, respectively. No TS Bases changes are directly tied to the MUR-PU TS changes.

As Attachments 5 through 8 contain information proprietary to Cameron, affidavits are included and signed by Cameron (a.k.a. Caldon), the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the NRC and address with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390. Accordingly, it is requested that the information that is proprietary to Cameron be withheld from public disclosure in accordance with 10 CFR 2.390. Correspondence with respect to the copyright or proprietary aspects of the information listed above or the supporting Cameron affidavits should reference Cameron letter CAW 19-01 through 04, as appropriate, and should be addressed to Joanna Phillips, Nuclear Sales Manager, Caldon Ultrasonics Technology Center, Cameron, 1000 McClaren Woods Drive, Coraopolis, PA, 15108.

Approval of the proposed amendment is requested by October 1, 2020 to support the Fall 2020 refueling outage for Unit 2. The proposed changes will be implemented within 180 days of the completion of the Unit 2 refueling outage scheduled for October-November 2020 and within 180 days of the completion of the Unit 1 refueling outage scheduled for March-April 2021.

Note that the spent fuel pool criticality analysis license amendment request (SNC letter NL-19-0796) has been submitted in advance of this MUR-PU LAR as previously discussed with the Farley NRC Project Manager and the NRC staff.

In accordance with 10 CFR 50.91, SNC is notifying the state of Alabama of this LAR by transmitting a copy of this letter to the designated state official.

If you have any questions or if additional information is needed, please contact Jamie Coleman at (205) 992.6611.

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

Executed on the 30th day of October 2019.

C.A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company

Enclosure: Description and Assessment of the Proposed Changes
Attachment 1: Operating License and Technical Specification Page Markups
Attachment 2: Retyped (Clean) Operating License and Technical Specification Pages
Attachment 3: Technical Information to Address the Application of WCAP-18124-NP-A in a non-Beltline Region
Attachment 4: Response to RIS 2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002
Attachment 5: Engineering Report: ER-1180P/NP Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM✓+ System" & Caldon Application for Withholding Proprietary Information from Public Disclosure CAW 19-01, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice
Attachment 6: Engineering Report: ER-1181P/NP Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM✓+ System" & Caldon Application for Withholding Proprietary Information from Public Disclosure CAW 19-02, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice
Attachment 7: Engineering Report: ER-1182P/NP Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 1" & Caldon Application for Withholding Proprietary Information from Public Disclosure CAW 19-03, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice
Attachment 8: Engineering Report: ER-1183P/NP Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 2" & Caldon Application for Withholding Proprietary Information from Public Disclosure CAW 19-04, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

cc: NRC Regional Administrator
NRC NRR Project Manager – Farley 1&2
NRC Senior Resident Inspector – Farley 1 & 2
Alabama - State Health Officer for the Department of Public Health
SNC Document Control R-Type: CFA04.054

ENCLOSURE

Description and Assessment of the Proposed Changes

Subject: Joseph M. Farley Nuclear Plant – Units 1 and 2 Submittal of License Amendment Request for Measurement Uncertainty Recapture Power Uprate

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ATTACHMENTS

Attachment 1: Operating License and Technical Specification Page Markups
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1 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, SNC requests an amendment to FNP Unit 1 Renewed Facility Operating License (NPF-2) and Unit 2 Renewed Facility Operating License (NPF-8) to increase the authorized core power level from 2775 MWt to 2821 MWt; an increase of approximately 1.7 percent RTP. Specifically, the proposed changes to OL 2.C(1) and TS Sections 1.1, 2.1.1, 3.4.1, and 5.6.6 are requested to allow for an MUR-PU.

In addition, SNC requests plant specific approval to apply WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET," in a limited application to predict fluence for non-beltline reactor vessel material.

2 DETAILED DESCRIPTION

2.1 System Design and Operation

Farley Units 1 and 2 are presently licensed for a core power rating of 2775 MWt. Using more accurate feedwater flow measurement instrumentation, SNC is seeking to increase the licensed core power to 2821 MWt.

The core power uprate for Farley Units 1 and 2 (hereby referred to as the MUR-PU) is based on recapturing measurement uncertainty currently included in the analytical margin originally required for emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K (ECCS evaluation models).

The 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 the 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 0.3 percent, SNC proposes to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K by approximately 1.66 percent (hereafter referred to as 1.7 percent – refer to Section 3.1).

The impact of the MUR-PU has been evaluated on the plant systems, structures, components, safety analyses, and off-site interfaces. Attachment 4 to this Enclosure summarizes these evaluations, analyses, and conclusions following the format in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002.

2.2 Current Technical Specifications Requirements

SNC proposes changes to the TS that are consistent with those approved for preceding Standard TS licensees' MUR-PU LARs. In addition, the peak fuel centerline temperature is revised to reflect the application of Topical Report (TR) WCAP-17642-P-A, Revision 1, Westinghouse Performance Analysis and Design Model (PAD5).

2.3 Reason for the Proposed Change

The proposed change is requested to take advantage of the incremental power increase allowed by the Appendix K rule change described in Section 2.1.

2.4 Description of the Proposed Change
The proposed changes to the OLs and TS are described below, with marked-up pages included in Attachment 1.

Operating License Maximum Power Level
Item 2.C(1), "Maximum Power Level," of the current OLs for FNP Units 1 and 2 (Renewed Facility Operating License Numbers NPF-2 and NPF-8); states: "Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal." This value is being increased to "2821 megawatts thermal."

TS Section 1.1, Definition of "Rated Thermal Power (RTP)"
The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 2775 MWt to 2821 MWt.

TS Section 2.1.1, "Reactor Core SLs"
TS 2.1.1.2 currently states: "In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU." TS 2.1.1.2 is being revised to state: "In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 9°F per 10,000 MWD/MTU."

TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
LCO 3.4.1 currently states: "RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR. LCO 3.4.1 is being revised to "RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 258,000 GPM, and ≥ the limit specified in the COLR."

TS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)"
TS 5.6.6.b currently states: "The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,' May 2004." TS 5.6.6.b is being revised to add: "and WCAP-18124-NP-A, Revision 0, 'Fluence Determination with RAPTOR-M3G and FERRET,' July 2018."

3 TECHNICAL EVALUATION

3.1 EVALUATION of MUR-PU - Summary

Farley Units 1 and 2 are presently licensed for an RTP of 2775 MWt. A more accurate feedwater flow measurement supports an increase to 2821 MWt. The technical evaluation for this MUR-PU addressed the following aspects:

1. The feedwater flow measurement technique and power measurement uncertainty
2. Accidents and transients that remain bounded at the higher power level, accidents and transients that are not bounded at the higher power level
3. Mechanical/structural/material component integrity and design

4. Electrical equipment design
5. System design
6. Operating, emergency, and abnormal procedures including associated operator actions
7. Environmental impact
8. Any changes to the TS including protective system setpoints

The evaluation conclusions are summarized in Attachment 4, in the format of RIS 2002-03 (Reference 1).

Cameron's bounding uncertainty analysis uses the plant-specific uncertainty data and combines these values with additional uncertainty terms related to the LEFM. Attachments 5 and 6 provide a complete list of the uncertainty terms that contribute to the total RTP uncertainty. The total RTP uncertainty for each unit is shown below and assumes the LEFM is operating in the "normal" mode.

Farley Unit 1: ± 0.32%
Farley Unit 2: ± 0.34%

In order to maintain consistency in the licensed thermal power for both units, the bounding uncertainty value is selected and used in the computations as follows. Based on the results, a bounding uncertainty value of ± 0.34% (from Farley Unit 2) will be used for both units.

The current licensed thermal power for each of the units at Farley is 2775 MWt.

- When factoring in the Cameron Unit 1 thermal power uncertainty of .32% the core power = 2775 MWt x (1+(.02-.0032)) = 2,821.62 MWt
- When factoring in the Cameron Unit 2 thermal power uncertainty of .34% the core power = 2775 MWt x (1+(.02-.0034)) = 2,821.065 MWt

The revised value of RTP is, therefore, requested to be 2821 MWt for both units at Farley.

Section 3.2 presents the nuclear steam supply system (NSSS) design thermal and hydraulic parameters used in support of the requested increase in power level. These parameters serve as the basis for the NSSS analyses and evaluations. The reactor core thermal power and/or NSSS thermal power are used as inputs to most plant safety, component, and system analyses.

3.2 NSSS DESIGN PARAMETERS

Introduction

The NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. They provide the primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as the basis for all the NSSS analyses and evaluations. As a result of the MUR-PU program, the Farley Units 1 and 2 NSSS design parameters have been revised as shown in Table 3.2-1. Table 3.2-1 provides information for the six cases associated with the Farley Units 1 and 2 MUR-PU program. These parameters have been incorporated, as required, into the applicable NSSS and balance of plant (BOP) systems and components evaluations, as well as safety analyses, performed in support of the MUR-PU Program.

Input Parameters

The major input parameters used in the calculation of the six cases provided in Table 3.2-1 for the Farley Units 1 and 2 MUR-PU Program are summarized by the following:

1. A bounding reactor core power level of 2831 MWt (NSSS power of 2841 MWt) was used for the analyses. This represents a 2.0 percent increase from the current licensed power level of 2775 MWt core power and 2785 MWt NSSS power. It is a bounding assumption intended to envelope the anticipated increase in licensed power of about 1.7 percent to 2821 MWt core power.
2. No change to the thermal design flow (TDF) of 86,000 gpm/loop.
3. Westinghouse Model 54F replacement steam generators (RSGs).
4. Steam generator tube plugging (SGTP) values of 0, 15 and 20 percent were considered.
5. Reactor vessel average temperatures (T_{avg}) of 577.2°F and 567.2°F were used for the analyses. This is the same temperature range used in the current plant condition.
6. A feedwater temperature (T_{feed}) of 446.0°F was used for the analyses, which was an increase from the current 443.4°F.

Parameter Cases

Table 3.2-1 provides the NSSS design parameter cases generated and used as the basis for the Farley Units 1 and 2 MUR-PU Program. The cases are defined as follows:

Cases 1 to 3 in Table 3.2-1 represent parameters based on a maximum T_{avg} of 577.2°F. Case 1, which is based on an average 0 percent SGTP level, yields the maximum secondary side steam generator pressure and temperature. All primary side temperatures are identical for Cases 1 to 3 in Table 3.2-1. The data provided in Note 2 of Table 3.2-1 were used in those NSSS analyses and evaluations that require an absolute upper limit steam pressure. These more limiting secondary side data are based on the Case 1 parameters with an assumed steam generator fouling factor of zero.

Cases 4 to 6 in Table 3.2-1 represent parameters based on a minimum T_{avg} of 567.2°F. Case 6, which is based on an average 20 percent SGTP, yields the minimum secondary side steam pressure and temperature. All primary side temperatures are identical for Cases 4 to 6 in Table 3.2-1.

The six cases of NSSS design parameters identified in Table 3.2-1 were used as the basis for the NSSS and, in some cases, BOP analytical efforts. The appropriate design parameters were used for each NSSS analysis, based on the conditions that were most limiting for that analytical area.

Table 3.2-1 NSSS Design Parameters for Farley Units 1 and 2 MUR-PU Program

Thermal Design Parameters	**Case 1**	**Case 2**	**Case 3**	**Case 4**	**Case 5**	**Case 6**
NSSS Power, MWt	2841	2841	2841	2841	2841	2841
10^6 Btu/hr	9694	9694	9694	9694	9694	9694
Reactor Power, MWt	2831	2831	2831	2831	2831	2831
10^6 Btu/hr	9660	9660	9660	9660	9660	9660
Thermal Design Flow, gpm/loop	86,000	86,000	86,000	86,000	86,000	86,000
Reactor 10^6 lb/hr	98.3	98.3	98.3	98.3	98.3	98.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250	2250	2250
Core Bypass, %	7.1 (1)	7.1 (1)	7.1 (1)	7.1 (1)	7.1 (1)	7.1 (1)
Reactor Coolant Temperature, °F						
Core Outlet	618.9	618.9	618.9	609.6	609.6	609.6
Vessel Outlet	614.0 (2)	614.0 (2)	614.0 (2)	604.5	604.5	604.5
Core Average	582.0	582.0	582.0	571.8	571.8	571.8
Vessel Average	577.2	577.2	577.2	567.2	567.2	567.2
Vessel/Core Inlet	540.5	540.5	540.5	529.9 (2)	529.9 (2)	529.9 (2)
Steam Generator Outlet	540.2	540.2	540.2	529.6	529.6	529.6
Steam Generator						
Steam Outlet Temperature, °F	514.4	508.2	505.6	503.4	497.2	494.5
Steam Outlet Pressure, psia	773	733	716	702	664 (2)	648 (2)
Steam Outlet Flow, 10^6 lb/hr total	12.54	12.52	12.51	12.50	12.49	12.49
Feed Temperature, °F	446.0	446.0	446.0	446.0	446.0	446.0
Steam Outlet Moisture, % max.	0.10	0.10	0.10	0.10	0.10	0.10
Tube Plugging Level, %	0	15	20	0	15	20
Zero Load Temperature, °F	547	547	547	547	547	547
Hydraulic Design Parameters						
Mechanical Design Flow, gpm/loop	101,800					
Minimum Measured Flow, gpm total	273,900					

Notes:

1. Core bypass flow includes 2.0 percent due to thimble plugs removed and 0.6 percent due to intermediate flow mixers and upflow conversion.
2. The following plant operational limits apply: vessel inlet temperature ≥ 530.6°F; vessel outlet temperature ≤ 613.3°F; full load steam pressure ≥ 690 psia.

3.3 Evaluation of Changes to License and Technical Specifications

The proposed changes to the OLs and TS described in Section 2.0, "Detailed Description," are evaluated as follows:

Operating License Maximum Power Level
Item 2.C(1), "Maximum Power Level," of the current operating licenses for FNP Units 1 and 2 (Renewed Facility Operating License Numbers NPF-2 and NPF-8); states: "Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal." This value is being increased to "2821 megawatts thermal."

Evaluation
The proposed increase in RTP from 2775 to 2821 MWt in the operating license is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this amendment request.

TS Section 1.1, Definition of "Rated Thermal Power (RTP)"
The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 2775 to 2821 MWt.

Evaluation
The proposed increase in RTP from 2775 to 2821 MWt in the TS definitions is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this amendment request.

TS Section 2.1.1, "Reactor Core SLs"
TS 2.1.1.2 currently states: "In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU." TS 2.1.1.2 is being revised to state: "In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 9°F per 10,000 MWD/MTU.

Evaluation
The proposed amendments would revise the peak fuel centerline temperature to reflect the fuel centerline melt temperature specified in TR WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," dated November 2017 (ADAMS Accession No. ML17338A396 [non-proprietary version]). PAD5 generates the best estimate and upper bound fuel temperatures as a function of local burnup and power. The generated fuel temperatures in conjunction with the burnup dependent fuel melting temperature (described in the TS Section 2.1.1 markup) are used to determine a local power to melt as a function of burnup. The PAD5 generated power-to-melt has been confirmed for the MUR-PU conditions and will be further confirmed as part of the cycle-specific Reload Safety Analysis Checklist. Further information on the application of PAD5 may be found in Attachment 4, Section III, item 26 (RCCA Ejection) and item 34 (Fuel Evaluations).

TS 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"
LCO 3.4.1 currently states: "RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method, and ≥ the limit specified in the COLR. LCO 3.4.1 is being revised to "RCS DNB parameters for pressurizer pressure, RCS

average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 258,000 GPM, and ≥ the limit specified in the COLR."

Evaluation
The MUR-PU DNBR calculations that use the statistical treatment of measurement uncertainties are based on a minimum measured flow of 273,900 GPM compared to the value of 263,400 GPM used in many of the current DNB analyses of record (AORs). The higher core flow is consistent with the value in the COLRs for the current operating cycles in the Farley units for those DNB events that are limiting below the first mixing vane grid. The DNB analyses which do not use the statistical treatment of measurement uncertainties continue to use the TDF of 258,000 gpm.

To facilitate this change, TS 3.4.1 is being changed to reflect the guidance in WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," D.S. Huegel, et al., January 19, 1999 and TSTF-339, R2.

TS 5.6.6 "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)"
TS 5.6.6.b currently states: "The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, " 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,' May 2004." TS 5.6.6.b is being revised to add: "and WCAP-18124-NP-A, Revision 0, 'Fluence Determination with RAPTOR-M3G and FERRET,' July 2018."

As detailed in Attachment 3 of this Enclosure, WCAP-18124-NP-A has been applied beyond the scope of current NRC approval in the extended beltline region. As such, NRC approval is requested for this limited plant specific application.

Evaluation
As discussed in Section IV.1 of Attachment 4, WCAP-18124-P/NP-A has been utilized for the fluence determination, and as such, has been added to the list of NRC approved documents for the PTLR. This addition is in accordance with the NRC safety evaluation approving the WCAP.

4 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix K, "ECCS Evaluation Models," requires that ECCS evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 31, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

The revision to 10 CFR 50, Appendix K does not permit licensees to utilize a lower uncertainty and increase thermal power without NRC approval. 10 CFR 50.90 requires that licensees desiring to amend an operating license file an amendment with the NRC. RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Power Uprate Applications," (Reference

1) provides NRC guidance for the content of LARs involving power uprates based on measurement uncertainty recapture.

4.2 Precedence

1. SNC submittal MUR-PU of the Vogtle Electric Generating Plant (ML072470691), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment, dated February 27, 2008 (TAC Nos. MD6625 and 6626) (ML080350347).
2. Virginia Electric and Power Company submittal for MUR-PU of the Surry Power Station, Units 1 and 2 (ML100320264), which was reviewed and approved by the NRC through a Safety Evaluation and License Amendment, dated September 24, 2010 (TAC Nos. ME3293 and ME3294) (ML101750002).
3. Duke Energy submittal for MUR-PU of the Catawba Nuclear Station Units 1 and 2, dated June 23, 2014 (ML14176A109), as supplemented by letters dated August 26 and December 15, 2014, and January 22, April 23, and November 16, 2015 (ML14245A059, ML14353A024, ML15029A417, ML15117A010, and ML15324A0B9, respectively), which was reviewed and approved by the NRC on April 29, 2016 (ML16081A333).

4.3 No Significant Hazards Consideration Determination Analysis

In accordance with 10 CFR 50.90, SNC requests an amendment to the Joseph M. Farley Nuclear Plant (FNP) Unit 1 Renewed Facility Operating License (NPF-2) and Unit 2 Renewed Facility Operating License (NPF-8) to increase the authorized core power level from 2775 megawatts thermal (MWt) to 2821 MWt - an increase of approximately 1.7% RTP.

Specifically, the proposed changes to OL 2.C(1) and TS Sections 1.1, 2.1.1, 3.4.1, and 5.6.6 are requested to allow for a Measurement Uncertainty Recapture (MUR) Power Uprate (MUR-PU). The MUR-PU for FNP Units 1 and 2 is based on recapturing measurement uncertainty currently included in the analytical margin originally required for Emergency Core Cooling System (ECCS) evaluation models performed in accordance with 10 CFR 50, Appendix K. SNC proposes to use state-of-the-art leading-edge flow meters (LEFMs) to more precisely measure the feedwater flow, which is used to calculate reactor power. These more precise measurements will reduce the degree of uncertainty in the power level, which is used to predict the ability of the reactor to be safely shutdown under postulated accident conditions.

SNC has evaluated whether a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," discussed as follows:

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 (RTP) for Farley Units 1 and 2 from 2775 to 2821 MWt - an increase of approximately 1.7 percent RTP. Evaluations have shown that all structures, systems, and components are capable of performing their design function at the uprated power of 2821 MWt. A review of station accident analyses found that all acceptance criteria are still met at the uprated power of 2821 MWt.

The PAD5 methodology used for evaluating the proposed change to the fuel centerline melt temperature limit has been reviewed by the NRC and found to be appropriately conservative per the NRC's Final Safety Evaluation for WCAP-17642-P-A, Revision 1.

The proposed use of the LEFM, the PAD5 methodology, the fluence calculations in the extended beltline region, and the increase in required reactor coolant system (RCS) flow serve to facilitate operations at the uprated power level and have no impact on the probability or consequences of any accident previously evaluated.

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 Farley updated Final Safety Analysis Report (FSAR). As discussed in Attachment 4, Sections II.1.D.iii, (items 22 and 23), and III.1 (items 7, 9, 24 and 26), the current analyses of record included sufficient margin in the secondary steam mass environmental releases to bound the increased values applicable for MUR-PU operation. The acceptance criteria for radiological consequences continue to be met at the uprated power level.

The proposed change does not involve any change to the design or functional requirements of the safety and support systems. That is, the increased power level neither degrades the performance of, nor increases the challenges to any safety systems assumed to function in the plant safety analysis. While power level is an input to accident analyses, it is not an initiator of accidents. The proposed change does not affect any accident precursors and does not introduce any accident initiators. The proposed change does not impact the usefulness of the surveillance requirements in evaluating the operability of required systems and components.

Additionally, evaluation of the proposed TS changes demonstrates that the availability of equipment and systems required to prevent or mitigate the radiological consequences of an accident is not significantly affected. The impact on the systems is minimal, and the overall impact on the plant safety analysis is negligible.

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

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

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced because of the proposed change. The use of the LEFM measurement system has been analyzed for Farley and failures of the system will have no adverse effect on any safety-related system or any systems, structures, or components (SSCs) required for transient mitigation. Similarly, projections of fluence for reactor vessel material in the extended beltline region will have no adverse effect on any safety-related system or any SSCs required for transient mitigation. SSCs previously required for the mitigation of a transient continue to be capable of fulfilling their intended design functions. The proposed change has no adverse effect on any

safety-related system or component and does not change the performance or integrity of any safety-related system.

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

Therefore, the proposed 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.

The margins of safety associated with the power uprate are those pertaining to core thermal power. These include fuel cladding, RCS pressure boundary, and containment barriers. Although the proposed amendment increases the Farley operating power level, the units 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.

Analyses demonstrate that the current design basis continues to be met after the measurement uncertainty recapture power uprate. Components associated with the RCS pressure boundary structural integrity, including pressure-temperature limits 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 Farley safety analyses, including the design basis radiological accident dose calculations, bound the proposed power uprate.

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

Therefore, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5 ENVIRONMENTAL CONSIDERATION

SNC has evaluated this license amendment request (LAR) against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21 (refer to Section VII.5 of Attachment 4). SNC 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 following:

1. The amendment involves no significant hazard consideration as demonstrated in Section 4.3.

2. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite. The principal barriers to the release of radioactive materials are not modified or affected by this change and no significant increases in the amounts of any effluent that could be released offsite will occur as a result of this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure. Because the principal barriers to the release of radioactive materials are not modified or affected by this change, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this change.

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

6 REFERENCES

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

Attachment 1

Operating License and Technical Specification Page Markups

in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

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

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

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

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2775~~ 2821 megawatts (thermal). Prior to attaining the power level, Alabama Power Company shall complete the preoperational tests, startup tests and other items identified in Attachment 2 to this renewed license in the sequence specified. Attachment 2 is an integral part of this renewed license.

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

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

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

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

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

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

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of ~~2775~~ megawatts thermal.

2821

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 216 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144
(4) Deleted per Amendment 149
(5) Deleted per Amendment 144

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the Low Temperature Overpressure Protection System applicability temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2775~~ $\boxed{2821}$ MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
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 rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by ~~58~~°F per 10,000 MWD/MTU.

9

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

258,000 GPM

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ ~~263,400 GPM when using the precision heat balance method, ≥ 264,200 GPM when using the elbow tap method,~~ and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1.

------------------------------------NOTE------------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

8. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

(Methodology for LCO 3.1.3 - Moderator Temperature Coefficient.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates and the LTOP System applicability temperature, shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

and WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018

(continued)

Attachment 2

Retyped (Clean) Operating License and Technical Specification Pages

in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

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

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

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

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2821 megawatts (thermal). Prior to attaining the power level, Alabama Power Company shall complete the preoperational tests, startup tests and other items identified in Attachment 2 to this renewed license in the sequence specified. Attachment 2 is an integral part of this renewed license.

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

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

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

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

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

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

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

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2821 megawatts thermal.

(2) <u>Technical Specifications</u>

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

(3) Deleted per Amendment 144

(4) Deleted per Amendment 149

(5) Deleted per Amendment 144

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the Low Temperature Overpressure Protection System applicability temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2821 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
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 rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 9°F per 10,000 MWD/MTU.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 258,000 GPM and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1.

------------------------------------NOTE------------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

(Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

8. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997.

(Methodology for LCO 3.1.3 - Moderator Temperature Coefficient.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates and the LTOP System applicability temperature, shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004 and WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

(continued)

Attachment 3

Technical Information to Address the Application of WCAP-18124-NP-A in a non-Beltline Region

Justification for Using RAPTOR-M3G for Reactor Pressure Vessel Extended Beltline Materials at J. M. Farley Units 1 and 2

Introduction

RAPTOR-M3G has been used in the fast neutron (E > 1.0 MeV) fluence evaluation in support of measurement uncertainty recapture power uprate (MUR-PU) for J.M. Farley Units 1 and 2. Per Regulatory Issue Summary (RIS) 2014-11 (Reference 1), the reactor pressure vessel (RPV) nozzles and any RPV materials that exceed $1x10^{17}$ n/cm^2 with fast neutron (E > 1.0 MeV) at the end-of-license extension (EOLE) are to be evaluated for fracture toughness. The RPV materials evaluated for J.M. Farley Units 1 and 2 are listed in Table 1. Certain materials in this table were determined to have fluence values less than 1E+17 n/cm^{2}and therefore did not require specific evaluation with respect to neutron embrittlement and fracture toughness.

The active core height extends from -182.88 cm to +182.88 cm in the RAPTOR-M3G model that represents the axial extension of the traditional beltline region of the reactor vessel which, by definition, is the region that directly surrounds the effective height of the active core (Reference 2). When compared to the axial elevations of the RPV materials evaluated in Table 1, the intermediate shell, intermediate shell longitudinal welds, intermediate shell to lower shell circumferential weld, lower shell, and lower shell longitudinal welds are categorized under the traditional beltline region. Therefore, these materials have been approved by the Nuclear Regulatory Commission (NRC) for generic application of RAPTOR-M3G for fast neutron (E > 1.0 MeV) fluence determination per WCAP-18124-NP-A (Reference 3).

The 54 effective full power years (EFPY) fast neutron fluence at the EOLE and locations for the RPV materials in the extended beltline region for J.M. Farley Units 1 and 2 are listed in Table 2. The limitations and conditions for using RAPTOR-M3G for fast neutron fluence determination are stipulated in Section 4.0 of the safety evaluation letter captured in Reference 3. The limitations and conditions of the safety evaluation for Reference 3 are quoted below:

1. *Applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (e.g., pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.*
2. *Least squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.*

The second limitation and condition listed above does not apply as the least-squares procedures were not used to adjust the calculated fast neutron (E > 1.0 MeV) fluence values for RPV materials evaluated in the reactor vessel integrity analysis. The least-squares results were only used to compare the calculations and measurements from the evaluated dosimetry and validate the neutron transport models, and those comparisons showed satisfactory results.

The Reference 3 neutron fluence methodology, however, has been used to determine the fast neutron (E > 1.0 MeV) fluence values for RPV materials in the extended beltline region. Therefore, use of the methodology in this region requires additional justification. The following information provides the

additional justification of using RAPTOR-M3G for fast neutron fluence determination for RPV extended beltline regions at J.M. Farley Units 1 and 2 by summarizing the additional benchmark data from a Westinghouse 4-loop pressurized water reactor (PWR).

Table 1: Locations for Reactor Vessel Materials

Material	Axial Location[1] [cm]	Azimuthal Location[2] [Degrees]
Inlet Nozzle at the Postulated 1/4T Flaw Axial Height	265.0	0.0 to 360.0
Outlet Nozzle to Nozzle Shell Welds - Lowest Extent		
Nozzle 1	259.64	25.0
Nozzle 2	259.64	145.0
Nozzle 3	259.64	265.0
Inlet Nozzle to Nozzle Shell Welds - Lowest Extent		
Nozzle 1	253.93	95.0
Nozzle 2	253.93	215.0
Nozzle 3	253.93	335.0
Nozzle Shell	206.78 to 448.54	0.0 to 360.0
Nozzle Shell to Intermediate Shell Circumferential Weld	204.88 to 206.78	0.0 to 360.0
Intermediate Shell		
Plate 1	-47.93 to 204.88	45.0 to 225.0
Plate 2	-47.93 to 204.88	225.0 to 45.0
Intermediate Shell Longitudinal Welds		
Weld 1	-47.93 to 204.88	45.0
Weld 2	-47.93 to 204.88	225.0
Intermediate Shell to Lower Shell Circumferential Weld	-51.11 to -47.93	0.0 to 360.0
Lower Shell		
Plate 1	-303.44 to -51.11	135.0 to 305.0
Plate 2	-303.44 to -51.11	305.0 to 135.0
Lower Shell Longitudinal Welds		
Weld 1	-303.44 to -51.11	135.0
Weld 2	-303.44 to -51.11	315.0
Lower Shell to Lower Vessel Head Circumferential Weld	-306.93 to -303.44	0.0 to 360.0

Notes:

1. Values listed are indexed to Z = 0.0 at the midplane of the active fuel stack.
2. Azimuthal locations are indexed to θ = 0.0 as shown on the reactor pressure vessel general assembly drawing.

Table 2: Fast Neutron Fluence at 54 EFPY and Locations for RPV Extended Beltline Region Materials

Material	Axial Location[1] [cm]	Unit 1 [n/cm^2]	Unit 2 [n/cm^2]
Inlet Nozzle at the Postulated 1/4T Flaw Axial Height	265.0	6.21E+16	6.49E+16
Outlet Nozzle to Nozzle Shell Welds - Lowest Extent			
Nozzle 1	259.64	5.60E+16	5.89E+16
Nozzle 2	259.64	4.71E+16	4.94E+16
Nozzle 3	259.64	9.48E+16	9.88E+16
Inlet Nozzle to Nozzle Shell Welds - Lowest Extent			
Nozzle 1	253.93	1.57E+17	1.64E+17
Nozzle 2	253.93	7.16E+16	7.49E+16
Nozzle 3	253.93	8.67E+16	9.08E+16
Nozzle Shell	206.78	8.20E+18	850E+18
Nozzle Shell to Intermediate Shell Circumferential Weld	204.88	9.25E+18	9.58E+18
Lower Shell to Lower Vessel Head Circumferential Weld	-303.44	1.71E+16	1.74E+16

Notes:

1. Values listed are indexed to Z = 0.0 at the midplane of the active fuel stack and only the closest distance to the core midplane is listed.

Additional Benchmarking Measurements

In order to collect measurement benchmark data for the extended beltline region, two sets of ex-vessel neutron dosimetry (EVND) have been installed at the elevation of the reactor vessel support for a Westinghouse 4-loop plant. The elevation of the reactor vessel support is approximately 8.5 feet above the core midplane. The specific axial locations of the EVND capsules to the core midplane (Z = 0.0 cm) and time of irradiation are listed in Table 3. The dosimeter foils included in these EVND capsules are listed in Table 4. The measured dosimetry reactions for those foils are listed in Table 5.

Table 3: Location and Time of Irradiation for Sensor Sets Analyzed at RPV Supports

Capsule ID	Sensor Location	Azimuthal Location	Axial Elevation [cm]	Cycle(s) of Irradiation
E	Ex-Vessel	180°	257.99	11
A	Ex-Vessel	225°	255.75	11
K	Ex-Vessel	180°	257.99	12 – 19

Table 4: Foil Sensor Set Contents in EVND at RPV Supports

Capsule ID	Radiometric Monitor Foils							
	Fe	Ni	Cu	Ti	Co	Nb	U-238	Np-237
E	x	x	x	x	x		x	x
A	x	x	x	x	x		x	x
K	x	x	x	x	x	x		

Table 5: Measured Dosimetry Reactions in EVND at RPV Supports

Material	Reaction of Interest	Neutron Energy Response(1)	Product Half-Life(2)
Copper	^{63}Cu (n,α) ^{60}Co	4.53-11.0 MeV	5.271 y
Titanium	^{46}Ti (n,p) ^{46}Sc	3.70-9.43 MeV	83.788 d
Iron	^{54}Fe (n,p) ^{54}Mn	2.27-7.54 MeV	312.13 d
Nickel	^{58}Ni (n,p) ^{58}Co	1.98-7.51 MeV	70.86 d
^{238}U	^{238}U (n,f) ^{137}Cs	1.44-6.69 MeV	30.05 y
Niobium	^{93}Nb (n,n') ^{93m}Nb	0.95-5.79 MeV	16.13 y
^{237}Np	^{237}Np (n,f) ^{137}Cs	0.68-5.61 MeV	30.05 y
Cobalt - Al	^{59}Co (n,g) ^{60}Co	Thermal	5.271 y

Note(s):

(1) Energies between which 90% of activity is produced (^{235}U fission spectrum). Ref. ASTM E844-18.

(2) Half-life data is from ASTM E1005-16.

Additional Benchmarking Neutron Transport Calculations

The neutron transport calculations for the additional benchmarking at the 4-loop Westinghouse plant extended beltline region followed the Westinghouse fluence methodology described in Reference 3, which is the same methodology used for the neutron transport calculations performed in support of J.M. Farley Units 1 and 2 MUR-PU work. In the application of this methodology to the fast neutron exposure evaluations for the 4-loop Westinghouse plant EVND dosimetry sets at the RPV supports, forward transport calculations were carried out to directly solve for the space- and energy-dependent scalar flux, $\varphi(r,\theta,z,E)$.

For the additional benchmark analysis, all of the transport calculations were carried out using the RAPTOR-M3G three-dimensional discrete ordinates code and the BUGLE-96 (Reference 4) cross-section library. The BUGLE-96 library provides a 67-group coupled neutron-gamma ray group cross-section data set produced specifically for Light Water Reactor (LWR) applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{20} order of angular quadrature.

A plan view of the reactor model is shown in Figure 1. In addition to the core, reactor internals, RPV, and concrete bioshield, the model also included explicit representations of the surveillance capsules, RPV clad, and RPV nozzles and supports. Section views of the reactor model are shown in Figure 2 and Figure 3.

In developing the model of the reactor geometry, nominal design dimensions were used for the various structural components. Water temperatures (and densities) in the core, bypass, and downcomer regions of the reactor were taken to be representative of full-power operating conditions. These coolant temperatures were varied on a cycle-specific basis. The reactor core itself was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel assembly grids, guide tubes, etc.

The r,θ,z geometric mesh description of the reactor model consisted of 241 radial by 190 azimuthal by 469 axial mesh intervals. Mesh sizes were chosen to ensure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration flux convergence criterion used in the calculations was 0.001.

The core power distributions used in the plant-specific transport analysis included fuel-assembly-specific initial enrichments, burnups, and axial power distributions. This information was used to develop spatial- and energy-dependent core source distributions averaged over each individual fuel cycle. Therefore, the results from the neutron transport calculations provided data in terms of the fuel cycle-averaged neutron fluence rate, which, when multiplied by the appropriate fuel cycle length, provide the incremental fast neutron fluence exposure for each fuel cycle. The energy distribution of the source was based on an appropriate fission split for uranium and plutonium isotopes based on the initial ^{235}U enrichment and burnup history of the individual fuel assemblies. From the assembly-dependent fission splits, composite values of energy release per fission, neutron yield per fission, and fission spectrum were determined. These fuel-assembly-specific neutron source strengths derived from the detailed isotopics were then converted from fuel pin Cartesian coordinates to the r,θ,z spatial mesh arrays used in the RAPTOR-M3G discrete ordinates calculations.

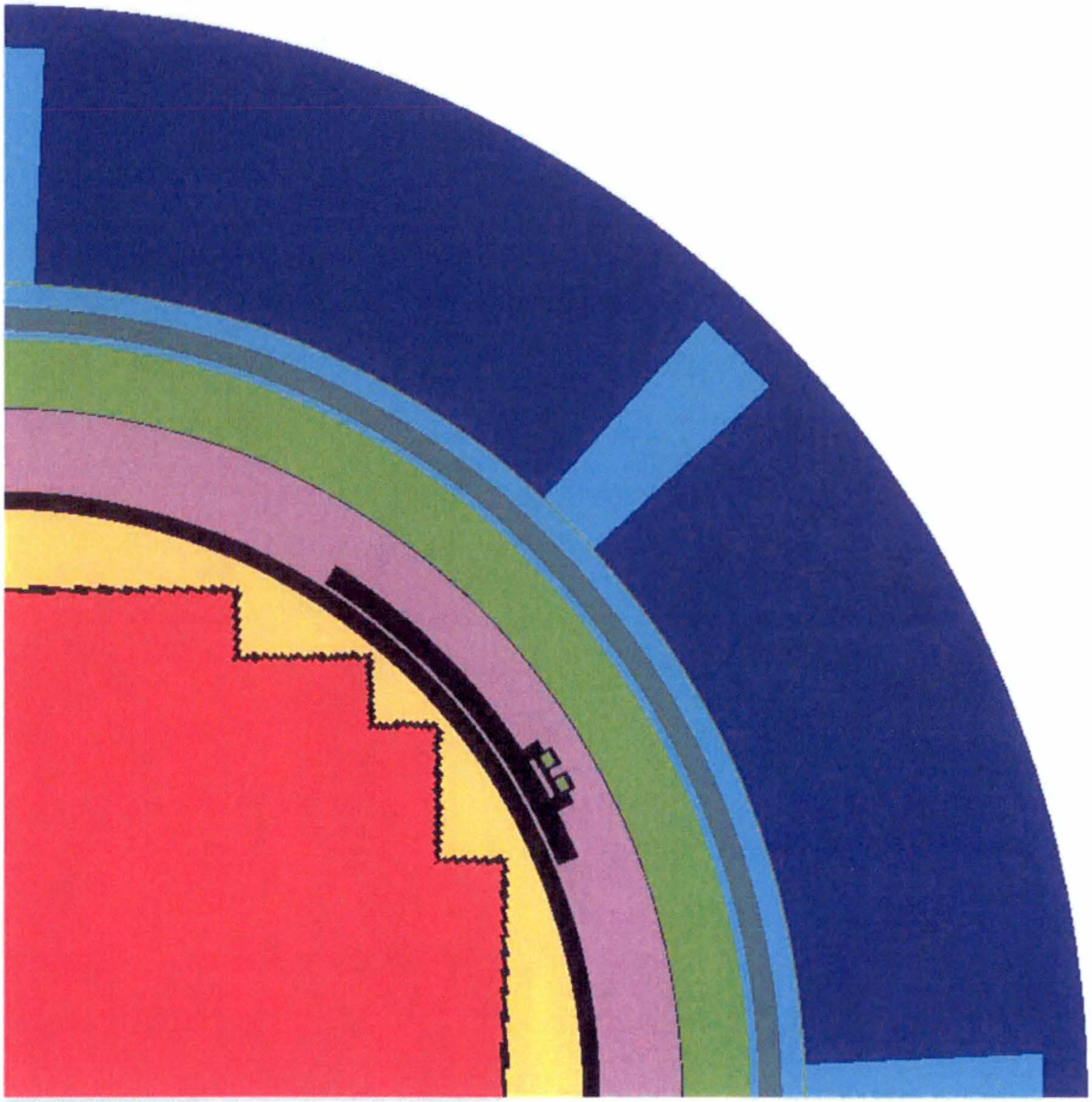

Figure 1: Reactor Geometry – Plan View at Core Midplane

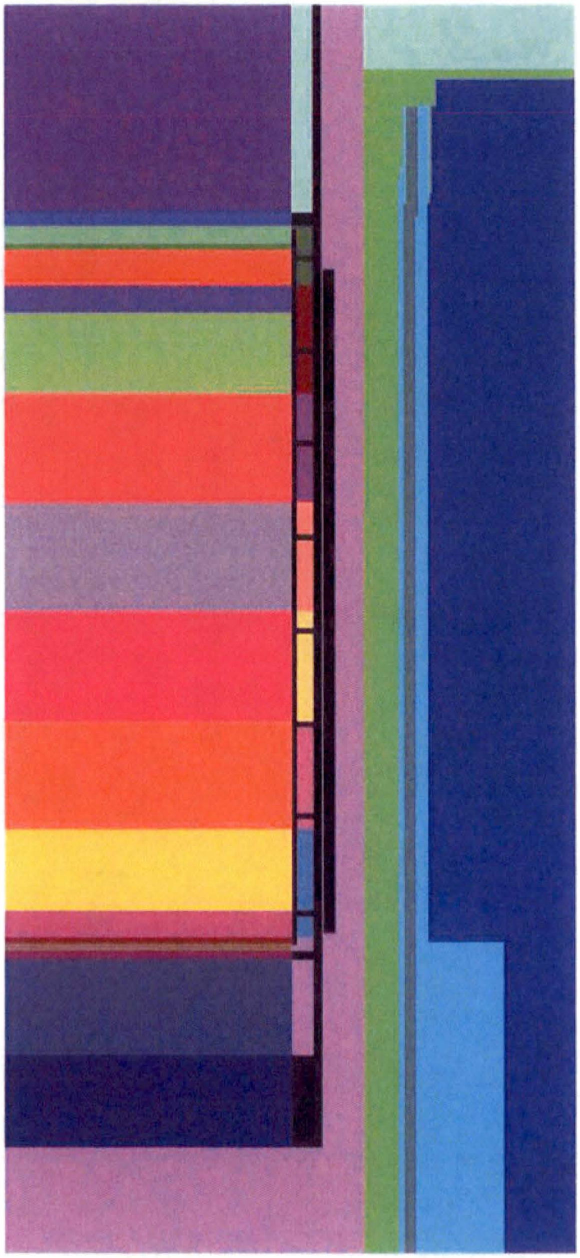

Figure 2: Reactor Geometry – Section View at Outlet Nozzle Centerline

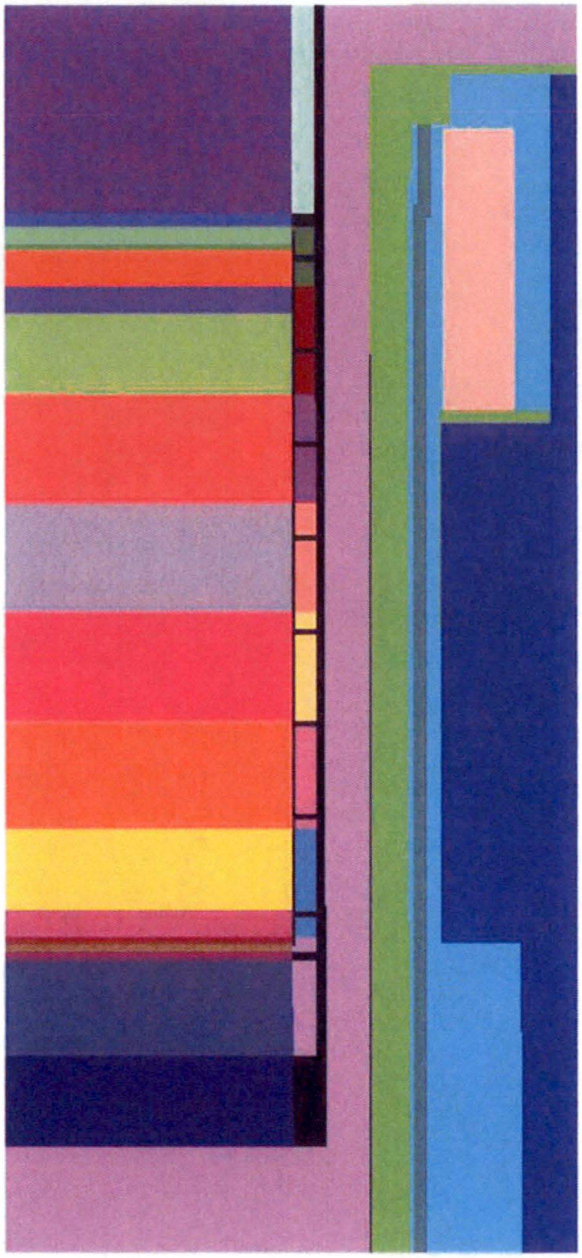

Figure 3: Reactor Geometry – Section View at Inlet Nozzle Centerline

Additional Benchmarking Dosimetry Evaluations

The evaluations of the neutron sensor sets contained in the EVND dosimetry capsules at the 4-loop Westinghouse plant RPV supports followed the state-of-the-art least-squares dosimetry evaluation methodology described in Section 3.0 of Reference 3.

Least-squares adjustment methods provide the capability of combining the measurement data with the neutron transport calculation resulting in a best-estimate neutron energy spectrum with associated uncertainties. Best-estimates for key exposure parameters such as fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate along with their uncertainties are then easily obtained from the adjusted spectrum. In general, the least-squares methods, as applied to reactor dosimetry evaluations, act to reconcile the measured sensor reaction rate data, dosimetry reaction cross sections, and the calculated neutron energy spectrum within their respective uncertainties.

For example,

$$R_i \pm \delta_{R_i} = \sum_g (\sigma_{ig} \pm \delta_{\sigma_{ig}})(\phi_g \pm \delta_{\phi_g})$$

relates a set of measured reaction rates, R_i, to a single neutron spectrum, φ_g, through the multigroup dosimeter reaction cross section, σ_{ig}, each with an uncertainty δ. The primary objective of the least-squares evaluation is to produce unbiased estimates of the neutron exposure parameters at the location of the measurement.

For the least-squares evaluation of the dosimetry, the FERRET code (Reference 5) was employed to combine the results of the plant-specific neutron transport calculations and sensor set reaction rate measurements to determine best-estimate values of exposure parameters along with associated uncertainties.

The application of the least-squares methodology requires the following input.

1. The calculated neutron energy spectrum and associated uncertainties at the measurement location.
2. The measured reaction rate and associated uncertainty for each sensor contained in the multiple foil set.
3. The energy-dependent dosimetry reaction cross sections and associated uncertainties for each sensor contained in the multiple foil sensor set.

For the current application, the calculated neutron spectrum at each measurement location was obtained from the results of the previously-described additional benchmarking neutron transport calculations. The spectrum at each sensor set location was input in an absolute sense (rather than simply a relative spectral shape). Therefore, within the constraints of the assigned uncertainties, the calculated data were treated equally with the measurements. The sensor reaction rates were derived from the measured specific activities of each sensor set and the operating history of the respective fuel cycles. The dosimetry reaction cross sections were obtained from the SNLRML dosimetry cross section library (Reference 6).

In addition to the magnitude of the calculated neutron spectra, the measured sensor set reaction rates, and the dosimeter set reaction cross sections, the least-squares procedure requires uncertainty estimates for each of these input parameters. The following provides a summary of the uncertainties associated with the least-squares evaluation of the dosimetry.

Additional Benchmarking Reaction Rate Uncertainties

The overall uncertainty associated with the measured reaction rates includes components due to the basic measurement process, the irradiation history corrections, and the corrections for competing reactions. A high level of accuracy in the reaction rate determinations is assured by utilizing laboratory procedures that conform to the ASTM International consensus standards for reaction rate determinations for each sensor type.

After combining all of these uncertainty components, the sensor reaction rates derived from the counting and data evaluation procedures were assigned the following net uncertainties for input into the least-squares evaluation:

Reaction	Uncertainty
$^{63}Cu\ (n,\alpha)\ ^{60}Co$	5%
$^{46}Ti\ (n,p)\ ^{46}Sc$	5%
$^{54}Fe\ (n,p)\ ^{54}Mn$	5%
$^{58}Ni\ (n,p)\ ^{58}Co$	5%
$^{238}U\ (n,f)\ ^{137}Cs$	10%
$^{93}Nb\ (n,n')^{93m}Nb$	10%
$^{237}Np\ (n,f)\ ^{137}Cs$	10%
$^{59}Co\ (n,\gamma)\ ^{60}Co$	35%*

These uncertainties are given at the 1σ level.

Additional Benchmarking Dosimetry Cross Section Uncertainties

As previously noted, the reaction rate cross sections used in the least-squares evaluations were taken from the SNLRML library. This data library provides reaction cross sections and associated uncertainties, including covariances, for 66 dosimetry sensors in common use. Both cross sections and uncertainties are provided in a fine multi-group structure for use in least-squares adjustment applications. These cross sections were compiled from the ENDF/B-VI cross section evaluations and have been tested with respect to their accuracy and consistency for least-squares evaluations. Further, the library has been empirically tested for use in fission spectra determination as well as in the fluence and energy characterization of 14 MeV neutron sources. Detailed discussions of the contents of the SNLRML library along with the evaluation process for each of the sensors are provided in Reference 6.

For sensors included in the dosimetry sets, the following uncertainties in the fission spectrum-averaged cross sections are provided in the SNLRML documentation package:

* The cobalt content of older Co-Al foils used in EVND is not known for certain, but is believed to be between 0.438% and 0.562%. To account for this unknown, the uncertainty assigned in the least-squares evaluations (typically 5%) was increased by roughly (0.562 / 0.438 = 1.28) to 5% + 28% = 33%. Rounded to the nearest five, an uncertainty of 35% was input.

Reaction	Uncertainty
^{63}Cu (n,α) ^{60}Co	4.08-4.16%
^{46}Ti (n,p) ^{46}Sc	4.51-4.87%
^{54}Fe (n,p) ^{54}Mn	3.05-3.11%
^{58}Ni (n,p) ^{58}Co	4.49-4.56%
^{238}U (n,f) ^{137}Cs	0.54-0.64%
^{93}Nb (n,n')^{93m}Nb	6.96-7.23%
^{237}Np (n,f) ^{137}Cs	10.32-10.97%
^{59}Co (n,γ) ^{60}Co	0.76-3.59%

These tabulated ranges provide an indication of the dosimetry cross section uncertainties associated with the sensor sets used in LWR irradiations.

Additional Benchmarking Calculated Neutron Spectrum Uncertainties

While the uncertainties associated with the reaction rates were obtained from the measurement procedures and counting benchmarks, and the dosimetry cross section uncertainties were supplied directly with the SNLRML library, the uncertainty matrix for the calculated spectrum was constructed from the following relationship:

$$M_{gg'} = R_n^2 + R_g * R_{g'} * P_{gg'}$$

Where R_n specifies an overall fractional normalization uncertainty and the fractional uncertainties $R_{g'}$ and R_g specify additional random groupwise uncertainties that are correlated with a correlation matrix given by:

$$P_{gg'} = [1-\theta]\delta_{gg'} + \theta\, e^{-H}$$

Where:

$$H = \frac{(g-g')^2}{2\gamma^2}$$

The first term in the correlation matrix equation specifies purely random uncertainties, while the second term describes the short-range correlations over a group range γ (θ specifies the strength of the latter term). The value of δ is 1.0 when g = g' and 0.0 otherwise.

The set of parameters defining the input covariance matrix for calculated spectra was as follows:

Flux Normalization Uncertainty (R_n)	15%
Flux Group Uncertainties (R_g, $R_{g'}$)	
(E > 0.0055 MeV)	15%

(0.68 eV < E < 0.0055 MeV)	25%
(E < 0.68 eV)	50%
Short-Range Correlation (θ)	
(E > 0.0055 MeV)	0.9
(0.68 eV < E < 0.0055 MeV)	0.5
(E < 0.68 eV)	0.5
Flux Group Correlation Range (γ)	
(E > 0.0055 MeV)	6
(0.68 eV < E < 0.0055 MeV)	3
(E < 0.68 eV)	2

These uncertainty assignments are consistent with an industry consensus uncertainty of 15-20% (1σ) for the fast neutron portion of the spectrum and provide for a reasonable increase in the uncertainty for neutrons in the intermediate and thermal energy ranges.

Additional Benchmarking Measurement-to-Calculation Comparison

The comparison of the measurement results from each of the sensor set irradiations at RPV supports with corresponding analytical predictions at the measurement locations are presented in Table 6 and Table 7. These comparisons are provided on two levels. On the first level, calculations of individual sensor reaction rates are compared directly with the measured data from the counting laboratories. This level of comparison is not impacted by the least-squares evaluations of the sensor sets. On the second level, calculated values of neutron exposure rates in terms of fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate are compared with the best-estimate exposure rates obtained from the least-squares evaluation.

In Table 6, comparisons of measurement-to-calculation (M/C) ratios are listed for the threshold sensors contained in the EVND dosimetry capsules irradiated at RPV supports that are approximately 8.5 feet above the core midplane. For the individual threshold foils, the average M/C ratio ranges from 0.62 to 1.28, with an overall average of 0.78 and an associated standard deviation of 25.5%. In this case, the overall average was based on an equal weighting of each of the sensor types with no adjustments made to account for the spectral coverage of the individual sensors.

In Table 7, best-estimate-to-calculation (BE/C) ratios for fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate resulting from the least-squares evaluation of the dosimetry sets is provided for the EVND capsules irradiated at the RPV supports, which are approximately 8.5 feet above the core midplane. The BE/C ratio for the fast neutron (E > 1.0 MeV) is 0.84 with an associated standard deviation of 8.9% and 0.93 with an associated standard deviation of 11% for the iron atom displacement (dpa). These BE/C ratios are within the ± 20% uncertainty at 1-σ level required by Regulatory Guide (RG) 1.190 (Reference 7).

In summary, for the extended beltline region, the M/C data provided in Table 6 as well as the BE/C data provided in Table 7 suggest that the calculations are over predicting the neutron exposure, in particular at the high end of the energy spectrum. For instance, the bottom of the 90% neutron response for the copper, titanium, iron, and nickel dosimeters is 4.53 MeV, 3.70 MeV, 2.27 MeV, and 1.98 MeV, respectively. Neutrons with energies greater than these constitute a small fraction of the neutron (E > 1.0 MeV) fluence rate in the extended beltline region. The BE/C values in Table 7 account for the spectral coverage of the different sensors, and provide an estimate of the key damage parameters, fast neutron (E > 1.0 MeV) fluence rate and iron atom displacement rate, that result from an uncertainty-weighted reconciliation of all of the measurements and calculations. The BE/C values

in Table 7 suggest that the calculated damage parameters are moderately conservative relative to the best-estimate values.

The uncertainty associated with a fluence determination methodology is comprised of two major components: the results of an analytic uncertainty analysis and the results of benchmarking comparisons. An analytic uncertainty analysis assesses the level of confidence in key input parameters to a fluence calculation and quantifies the impact that plausible input parameter variations have on calculated fluence results. Benchmarking comparisons refer to comparisons of fluence calculations performed with a candidate methodology to alternate calculations or to measurements from a representative environment.

The fast neutron (E > 1.0 MeV) fluence rate analytic uncertainty for the Westinghouse fluence methodology in the reactor cavity at locations opposite the top and bottom of the active fuel is 17–18% at the 1σ level (Reference 3). Combining this analytic uncertainty with the other uncertainty components identified in Reference 3 yields a net uncertainty (1σ) estimate of 19-20%.

A comprehensive analytic uncertainty assessment for the extended beltline region has not been completed. Preliminary estimates at this time suggest that the analytic uncertainty in this region will be moderately greater than the analytic uncertainty attributed to the cavity locations at the elevation of the top of the core. The benchmarking data set for the extended beltline region evaluated in this document is small (3 dosimetry capsules); however, it seems that, after combining the results in this document with the expected results of a comprehensive analytic uncertainty assessment, the net methodology uncertainty for the RAPTOR-M3G methodology may be on the order of 30% in the vicinity of the RPV supports.

Table 6: Measured-to-Calculated (M/C) Reaction Rates – Ex-Vessel Capsule Located in the Vicinity of the RPV Supports

Reaction	Capsule E	Capsule A	Capsule K	Average	% Std. Dev.
^{63}Cu (n,α) ^{60}Co	0.65	-	0.58	0.62	8.0
^{46}Ti (n,p) ^{46}Sc	0.73	0.65	0.67	0.68	6.1
^{54}Fe (n,p) ^{54}Mn	0.72	0.66	0.64	0.67	6.2
^{58}Ni (n,p) ^{58}Co	0.75	0.67	0.65	0.69	7.7
^{238}U(Cd) (n,f) ^{137}Cs	1.03	0.89	-	0.96	10.3
^{93}Nb (n,n') ^{93m}Nb	-	-	1.28	1.28	-
^{237}Np(Cd) (n,f) ^{137}Cs	1.12	0.80	-	0.96	23.6
Average of M/C Results				0.78	25.5

Table 7: Best-Estimate-to-Calculated (BE/C) Exposure Rates – Ex-Vessel Capsule Located in the Vicinity of the RPV Supports

Capsule	Neutron (E > 1.0 MeV) Fluence Rate BE/C	Iron Atom Displacement Rate BE/C
E	0.91	0.99
A	0.76	0.82
K	0.84	1.00
Average	0.84	0.93
% Std. Dev.	8.9	11

Justification Conclusions

Additional benchmarking work was performed at a 4-loop Westinghouse plant on RPV supports that are about 8.5 feet above the core midplane. This work concluded that the RAPTOR-M3G fluence determination methodology has about 30% uncertainty in the fast neutron (E > 1.0 MeV) determination and the calculations typically overestimate the fast neutron (E > 1.0 MeV) fluence and iron atom displacement (dpa) at the extended beltline region. Based on the J.M. Farley Units 1 and 2 reactor vessel integrity materials analysis, none of the extended beltline materials are limiting and these materials have significant margin before becoming the limiting materials.

The RPV extended beltline materials evaluated for J.M. Farley Units 1 and 2 in Table 2 are mostly located within an axial distance of 8.5 feet above or below the core midplane. The only exception is the lower shell to lower vessel head circumferential weld, which is approximately 1.5 feet (45 cm) further away. However, this circumferential weld has a calculated fast neutron (E > 1.0 MeV) fluence of 1.74E+16 n/cm^2, which is more than a factor of 5 lower than the prescribed threshold of 1E+17 n/cm^2 for the definition of the extended beltline region. Because the RAPTOR-M3G fluence determination methodology uncertainty only increases from 19 – 20 % at the top of the active fuel (+182.88 cm above core midplane) to approximately 30% at the RPV supports (+257.99 cm above the core midplane), it is not credible that the methodology uncertainty would increase from approximately 30% to 500% for an additional 1.5-foot (45-cm) axial distance from the core midplane, and thus require this circumferential weld to be included as extended beltline material that has to be evaluated for fracture toughness embrittlement effect.

The fast neutron (E > 1.0 MeV) fluence values reported at elevations in the proximity of the RPV supports in the additional benchmarking case may also be used to estimate the uncertainty for the inlet and outlet nozzles that are evaluated for fracture toughness due to structural discontinuities per Reference 1. Therefore, the methodology uncertainty for fluence determination for these RPV extended beltline materials is also approximately 30% and the calculated fast neutron (E > 1.0 MeV) fluence tends to be overestimated when compared to the measurement benchmark data. In addition, certain fast neutron (E > 1.0 MeV) fluence values used as input in the downstream fracture toughness evaluation have extra margin built-in. For example, the highest inlet nozzle to upper shell girth weld fast neutron (E > 1.0) fluence value has been used for both inlet and outlet nozzles to upper shell girth welds, respectively, during the evaluation as this approach is conservative because the inlet nozzle to upper shell girth weld is closer to the core midplane. Additionally, the reactor vessel integrity analysis showed that there is significant margin prior to these materials becoming limiting. Finally, Section 3.4 of Reference 8 states that unless the fast neutron (E > 1.0 MeV) fluence for the nozzle material is greater than 4.28E+17 n/cm^2, embrittlement need not be considered for nozzle forging evaluation and the nozzles will be non-limiting compared to the beltline with respect to the pressure-temperature limit curves. Embrittlement was conservatively considered in the reactor vessel integrity analyses for nozzle materials, even if the fluence was below this threshold. The fast neutron (E > 1.0 MeV) fluence values reported for both the inlet and outlet nozzles in Table 2 are more than a factor of 2 lower than this fast neutron (E > 1.0 MeV) fluence threshold. As the evaluated net RAPTOR-M3G methodology uncertainty is approximately 30% for this elevation based on additional benchmarking and preliminary analytical uncertainty analysis, it is also not credible that the inlet and outlet nozzle fast neutron (E > 1.0 MeV) fluence at the EOLE will exceed 4.28E+17 n/cm^2.

Therefore, based on the additional benchmarking at the RPV extended beltline region and margin assessment provided in Reference 8, the RAPTOR-M3G fluence determination methodology is justified to be applicable to J.M. Farley Units 1 and 2 RPV extended beltline region fast neutron (E > 1.0 MeV) fluence determination for fracture toughness evaluation.

Application of WCAP-18124-NP-A, Revision 0 to the Extended Beltline Region - Conclusion

Limitation and Condition #1 has been addressed in that the additional benchmarking at the RPV extended beltline region summarized herein, margin assessment documented in Reference 8, and the MUR-PU analysis have provided additional justification supporting the use of the Reference 3 methodology for the extended beltline regions of the J.M. Farley Units 1 and 2 RPVs.

References:

1. Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," October 14, 2014. (ML14149A165)
2. 10 CFR Appendix G to Part 50, "Fracture Toughness Requirements"
3. Westinghouse Report, WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018
4. RSICC Data Library Collection DLC-185, "BUGLE-96 Coupled 47 Neutron, 20 Gamma-Ray Group Cross Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," March 1996 (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory)
5. RSICC Computer Code Collection PSR-145 "FERRET: Least-Squares Solution to Nuclear Data and Reactor Physics Problems," January 1980 (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory)
6. RSICC Data Library Collection DLC-178, "SNLRML Recommended Dosimetry Cross Section Compendium," July 1994 (Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory)
7. U.S. Nuclear Regulatory Commission Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001
8. Pressurized Water Reactor Owners Group (PWROG) Report, PWROG-15109-NP, Revision 0, "PWR Pressure Vessel Nozzle Appendix G Evaluation," February 2018

Attachment 4

Response to RIS 2002-03, "Guidance on Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002

ATTACHMENT 4: SUMMARY OF RIS 2002-03 REQUESTED INFORMATION FOR FARLEY NUCLEAR PLANT

TABLE OF CONTENTS FOR ATTACHMENT 4

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 at Farley Units 1 and 2 is a Cameron (aka Caldon) leading edge flow meter (LEFM) CheckPlus ultrasonic multi-path transit time flowmeter as described in the following topical reports:

- Cameron Engineering Report ER-80P, Revision 0, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," March 1997, (Reference I.1)
- Cameron Engineering Report ER-157(P-A), Revision 8 and Revision 8 Errata, "Supplement to Cameron Topical Report ER-BOP, 'Basis for Power Uprates with an LEFM Check or a CheckPlus System,'" May 2008 (Reference I.2)

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

RESPONSE:

The Cameron LEFM check instruments (Report ER-80P) was reviewed and approved by the Nuclear Regulatory Commission (NRC) in the Safety Evaluation Report (SER) contained in Letter 1 that follows. Subsequently, the LEFM CheckPlus instruments (Report ER-157P-A), Revision 8, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System" was reviewed and approved by the NRC in the SER in Letter 2.

1. 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' (TAC Nos. MA2298 and MA2299)," March 8, 1999 (ADAMS Accession Number 9903190065, legacy library) (Reference I.3)
2. NRC letter from Thomas B. Blount, Deputy Director, NRC, to Mr. Ernest Hauser, Cameron, "Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, 'Caldon Ultrasonics Engineering Report ER-157P, Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System' (TAC No. ME1321)," August 16, 2010 (ML 102160663) (Reference I.4)

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 (UFM) system will be permanently installed in Farley Units 1 and 2 and operated in accordance with the manufacturer's requirements as described in References I.1 and I.2. The system will be used for continuous calorimetric power determination by direct links with the Farley Unit 1 and 2 Integrated Plant Computer. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis. Even though the LEFM CheckPlus system is not safety-related it is designed and manufactured in accordance with Cameron's Quality Assurance Program, which is certified to ISO 9001:2015 and supplemented by the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The LEFM spool pieces installation location in Farley, Units 1 and 2, is within the 14-inch diameter feedwater piping downstream of the common feedwater header where the lines split into three straight sections of piping going to the three steam generators (SGs). The spool pieces are installed sufficiently upstream of the existing feedwater flow venturis and downstream of any piping components such that no adverse interactions are created. No flow straighteners will be installed. The LEFM CheckPlus UFM system is composed of three metering sections for the feedwater lines (each metering section includes two electronic transmitters, two pressure transmitters, 16 acoustic transducers, RTDs, and a pressure port), and the system includes two Processing Units, and instrument cables for each transducer in the system. The LEFM metering sections and transmitters are located in the Turbine Building (TB), and the Processing Units are located in the computer room in the Auxiliary Building.

Each of the LEFM spool pieces were calibrated at the Alden Research Laboratory facility using a hydraulic duplicate of the principal hydraulic features of the plant configuration. The calibration tests determined the meter factor (i.e., meter calibration constant) for each of the Unit's LEFMs. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests were also performed at the Alden Research Laboratory facility to determine meter factor sensitivity to upstream hydraulics. Copies of the Unit specific Meter Factor Calculation and Accuracy Assessments (References I.7 through I.8) based on the Alden Laboratory test results are provided in Attachments 7 through 8 of this LAR. Note that Farley Unit 2 contains schedule 80 piping at the interface of the LEFM spool pieces for the A and B feedwater piping loops. The geometry of this deviation to the plant configurations used for the Unit 2 spool piece calibrations was evaluated by Cameron. This evaluation resulted in LEFM installation conditions that stay within the bounding uncertainty of the calibration.

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 Cameron Topical Report ER-80P (Reference I.1), the NRC established four criteria each licensee must address. The Southern Nuclear Operating Co. (SNC) response to those criteria for Farley, Units 1 and 2 is provided in Sections I.1.D.i through I.1.D.iv.

In approving Cameron Engineering Report 157-P (Reference I.1) the NRC established five additional criteria to be addressed by each licensee. SNC's response for how each of the additional criteria will be satisfied for Farley, Units 1 and 2 is provided in Sections I.1.D.v through I.1.D.ix.

I.1.D.i <u>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.</u>

RESPONSE:

Implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for continued calibration and maintenance of the LEFM system. Plant maintenance and calibration procedures will be revised to incorporate Cameron's maintenance and calibration requirements prior to raising power above the current licensed thermal power (CLTP) of 2775 MWt. The Farley Unit 1 and Unit 2 technical requirement manual (TRM) will be revised as discussed in Sections I.1.G and H below to address contingencies for non-functional LEFM instrumentation.

A modification package will be developed for each installation outlining the steps to install and test the LEFM CheckPlus system. When each Unit is shut down for their respective refueling outages (Farley Unit 2 2R27, October 2020 and Farley Unit 1 1R30, March 2021), the LEFM CheckPlus systems will be installed. Following installation, testing will include an in-service leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. The LEFM CheckPlus system installation and commissioning will be performed according to Cameron procedures. Commissioning and start-up of the LEFM CheckPlus system will be performed by qualified Cameron personnel with site personnel assistance. The commissioning process provides final positive confirmation that actual field performance meets the uncertainty bounds established for the instrumentation. Final site-specific uncertainty analyses acceptance will occur after completion of the commissioning process.

The Farley LEFM CheckPlus system was calibrated in a site-specific model test at Alden Research Laboratory. A copy of the Alden Research Laboratory certified calibration reports is contained in the Cameron Meter Factor Reports, References I.7 through I.8 (LAR Attachments 7 through 8, Appendix A.3). The testing at Alden Laboratory provides traceability to National Standards. The spool piece calibration factor uncertainty is based on these Cameron

engineering reports. The calibration tests included a site-specific model of each of the Units' hydraulic geometry. The installations at Farley do not require and will not employ upstream flow straighteners. A discussion of the impact of plant-specific installation factors on the feedwater flow measurement uncertainty is also provided in Attachments 7 through 8.

Preventive maintenance will be performed based on vendor recommendations. Required training materials to be developed and implemented will include training of operating and maintenance personnel. The preventive maintenance program and the LEFM CheckPlus system continuous self-monitoring feature ensure that the LEFM remains bounded by the Topical Report ER-80P (Reference I.1), as supplemented by ER-157P (Reference I.2), analysis and assumptions. Continued adherence to these requirements assures that the LEFM CheckPlus system is properly maintained and calibrated. The preventive maintenance activities will be identified via the associated plant modification package. Typical activities performed include power supply checks, pressure transmitter checks, and clock verifications. Maintenance of the LEFM system will be performed by personnel with appropriate knowledge of the LEFM.

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

RESPONSE:

Criterion 2 does not apply to Farley Unit 1 and Unit 2 as they do not have LEFMs installed at this time. Farley currently uses flow venturis to measure feedwater flow to support the secondary calorimetric power measurements. The LEFM CheckPlus system is scheduled to be installed in Farley Unit 2 in October 2020 (2R27) and in Farley Unit 1 in March 2021 (1R30). After the LEFM CheckPlus system is installed and operational, data will be collected comparing the LEFM CheckPlus operating data to the venturi data to verify consistency between thermal power calculation based on LEFM data and other plant parameters.

I.1.D.iii **Criterion 3 from ER-80P - Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation for comparison.**

RESPONSE:

The LEFM uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1 and Alden Research Laboratory Inc. calibration tests. This methodology has been used for instrument uncertainty calculations for multiple MUR power uprates with subsequent NRC approval.

Cameron has performed Unit specific bounding uncertainty analysis for Farley Unit 1 and Unit 2 (References I.5 and I.6). Copies of these analyses are provided in Attachments 5 through 6 of the LAR. The calculations in these analyses are consistent with Cameron's Topical Report ER-80P (Reference I.1), as supplemented by ER-157P (Reference I.2). These Topical Reports have been approved by the NRC in References I.3 and I.4. The core thermal power uncertainty calculation, which takes into account the uncertainty associated with the feedwater flow venturis is performed in accordance with WCAP-12771 (Reference I.9).

I.1.D.iv **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:

Criterion 4 does not apply to Farley Unit 1 and Unit 2. The calibration factors for the Farley Unit 1 and Unit 2 spool pieces were established by tests of these spools at Alden Research Laboratory. Cameron engineering reports for each of the Units, evaluating the calibration test data from Alden Research Laboratory, have been completed (References I.7 through I.8) and are provided in LAR Attachments 7 through 8 (Appendix A.3). The calibration factors used for each Units' LEFMs are based on the analysis contained in these reports. The uncertainties in the calibration factor for the spools are based on the Cameron site-specific engineering reports, (References I.5 and I.6) and are provided in LAR Attachments 5 through 6 (Appendix A.3).

Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies bounding calibration test data (see Appendix F of ER-80P (Reference I.1).

I.1.D.v **Criterion 1 from ER-157P, Rev 8 - Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.**

RESPONSE:

As described in Section I.1.G, operation above 2775 MWt will be limited to 72 hours if the LEFM CheckPlus is not functional.

I.1.D.vi **Criterion 2 from ER-157P, Rev 8 - A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.**

RESPONSE:

Farley will not apply a secondary condition with LEFM CheckPlus in a degraded condition with increased uncertainty. As described further in Section I.1.G, Farley will conservatively respond to a single path or single plane failure in the LEFM CheckPlus in the same manner as a complete system failure and operation above 2775 MWt will be limited to 72 hours.

I.1.D.vii **Criterion 3 from ER-157P, Rev 8 - An applicant with a comparable geometry can reference the above Section 3.2.1 finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with the use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.**

RESPONSE:

The downstream piping geometry concerns described in Section 3.2.1 of the NRC SE (Reference I-4) are not applicable to Farley. As discussed above, the LEFM spool pieces will be installed in straight sections of feedwater piping and have been tested at Alden Laboratories in hydraulically equivalent configurations. Additionally, as described in response to Qualification 2 above, Farley does not propose to apply a secondary condition to allow use of the LEFM CheckPlus with an increased uncertainty (in a "Check" equivalent mode).

I.1.D.viii **Criterion 4 from ER-157P, Rev 8 - An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17. (Reference 17 = Letter from Hauser, E (Cameron Measurement Systems), to U.S. Nuclear Regulatory Commission, "Documentation to support the review of ER-157P, Revision 8: Engineering Report ER-790, Revision 1, 'An Evaluation of the Impact of 55 Tube Permutit Flow Conditioners on the Meter Factor of an LEFM CheckPlus'," March 19, 2010) Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.**

RESPONSE:

The feedwater piping configurations at Farley do not necessitate or use upstream flow straighteners.

I.1.D.ix **Criterion 5 from ER-157P, Rev 8 - An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18. (Reference 18 = Letter from Hauser, E (Cameron Measurement Systems), to U.S. Nuclear Regulatory Commission, "Documentation to support the review of ER-157P, Revision 8: Engineering Report ER-764, Revision 0, 'The Effect of the Distribution of the Uncertainty in Steam Moisture Content on the Total Uncertainty in Thermal Power'," March 18, 2010)**

RESPONSE:

The uncertainty associated with steam moisture content for Farley Units 1 and 2 is based on a 0.1-percent moisture content in the steam. Farley's uncertainty in the moisture carryover is on the same order of magnitude as the other calorimetric uncertainties (±0.092 percent). As discussed in Engineering Report ER-764 (Reference I.11), this would not be classified as a "large uncertainty" in the moisture content. Therefore, Criterion 5 does not apply to Farley.

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:

Feedwater flow and temperature are the main inputs for determining the plant secondary calorimetric power, which is used in turn to determine the reactor thermal power. The feedwater mass flow rate and temperature are transmitted from the LEFM electronics cabinet to the Unit's plant process computer (PPC) for use in the calorimetric software application (secondary plant

heat balance) which determines reactor thermal power. The improved measurement accuracy for feedwater mass flow and temperature over that currently obtainable with venturi-based instrumentation and thermocouples reduces total uncertainty in the calculation of rated thermal power (RTP) to less than the nominal 2 percent currently assumed in many accident analyses, thereby allowing an increase in reactor thermal power equivalent to the decrease in uncertainty.

The Cameron uncertainty calculations for Farley Unit 1 and Unit 2 (Reference I.5 through I.6, respectively), are documented and provided in LAR Attachments 5 through 6. The key parameters and their uncertainty are summarized in Table I-1. In addition to the uncertainties associated with the parameters provided by the LEFM CheckPlus system, the uncertainties associated with the other plant parameters used by the plant computer to calculate the calorimetric are combined and taken into consideration. Acceptance testing following installation of the CheckPlus systems in Farley Unit 1 and Unit 2 will confirm that all uncertainty contributors are within the bounds of the mass flow error analysis.

The Farley uncertainties for the calorimetric inputs provided by the Cameron LEFM CheckPlus system are listed below. These uncertainty values were determined utilizing the calculation methodology described in Cameron Engineering Reports ER-80P and ER-157P (References I.1 and I.2).

- Farley Unit 1: Thermal power uncertainty using a Fully Functional LEFM CheckPlus system is [± 0.32 percent]
- Farley Unit 2: Thermal power uncertainty using a Fully Functional LEFM CheckPlus system is [± 0.34 percent]

Table I-1: Total Thermal Power Uncertainty Determination for Farley Units 1 and 2

Item	Parameter (1)	Unit 1 Uncertainty	Unit 2 Uncertainty
1	Hydraulics: Profile Factor	0.15%	0.19%
2	Geometry:		
	Spool Dimensions	0.12%	0.12%
3	Time Measurements		
	Time of Flight Measurements	0.04%	0.04%
	Non-fluid delay	0.03%	0.03%
4	Feedwater Density (2) (3)		
	Feedwater Density/Correlation	0.04%	0.04%
	Feedwater Density/Temperature	0.05%	0.05%
	Feedwater Density/Pressure	0.01%	0.01%
5	Subtotal: Mass Flow Uncertainty (Root Sum Square of Items 1-4)	0.26%	0.28%

Item	Parameter [1]	Unit 1 Uncertainty	Unit 2 Uncertainty
6	Feedwater Enthalpy[2] [3]		
	Feedwater Enthalpy/Temperature	0.08%	0.08%
	Feedwater Enthalpy/Pressure	0.00%	0.00%
	Power Uncertainty, Thermal Expansion	0.03%	0.03%
7	Steam Enthalpy:		
	Steam Enthalpy/Moisture	0.092%	0.092%
	Steam Enthalpy/Pressure	0.076%	0.076%
8	Gains/Losses	0.085%	0.085%
9	Total Thermal Power Uncertainty [4]	0.32%	0.34%

Notes:
1. Items 1 through 6 are directly associated with the Caldon LEFM CheckPlus system device. Items 7 and 8 are based on other plant process inputs.
2. The bounding uncertainties is based on a pressure input uncertainty of +/- 15 psi.
3. The bounding uncertainties for temperature is +/- 0.57°F.
4. Farley Unit 1 and 2 Power Uncertainty Analyses (References I.5 and I.6).

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 <u>Maintaining calibration</u>

RESPONSE:

Calibration and maintenance for the LEFM CheckPlus hardware and instrumentation will be performed using procedures based on the appropriate Cameron LEFM CheckPlus technical manuals, which ensures that the LEFM CheckPlus system remains bounded by the Topical Report ER-80P analysis and assumptions. Routine preventive maintenance activities for the LEFM will be as discussed in Section I.1.D.1. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work will be planned and executed in accordance with established Farley work control processes and procedures.

I.1.F.ii <u>Controlling software and hardware configuration</u>

RESPONSE:

Cameron's manufacturing and quality programs are certified to ANSI/ISO/ASQC9001 and supplemented by quality assurance criteria for Nuclear Power Plants defined in 10 CFR 50 Appendix B and 10CFR21 for Reporting of Defects and Nonconformance. Cameron software is

developed and maintained under a Verification and Validation (V&V) program consistent with ASME NQA-1a-1999 Subpart 2.7 (Reference I.10).

After installation, the LEFM CheckPlus system software configuration will be maintained using existing procedures and processes. The plant computer software configuration is maintained in accordance with the Farley change control process, which includes V&V of changes to software configuration. Configuration of the hardware associated with the LEFM CheckPlus system and the calorimetric process instrumentation will be maintained in accordance with Farley's configuration control processes.

I.1.F.iii Performing corrective actions

RESPONSE:

Plant instrumentation that affects the power calorimetric, including the LEFM inputs, will be monitored by Farley personnel. In accordance with the Farley corrective action program (CAP), deficiencies will be documented, and corrective actions will be identified and implemented.

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

RESPONSE:

Deficiencies associated with the vendor's processes or equipment will be reported in accordance with the Farley CAP.

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

RESPONSE:

The Farley Unit 1 and Unit 2 CheckPlus is under Cameron's V&V program, and procedures are maintained by Cameron for user notification of important deficiencies. The Farley Unit 1 and 2 Uncertainty Analyses (References I.5 and I.6) include an uncertainty value for transducer location, as described in Appendix D of ER157(P-A), Rev. 8 (Reference I.2). Farley also has existing processes for addressing manufacturer deficiency reports. Applicable deficiencies will be documented and addressed in the Farley CAP.

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

RESPONSE:

Farley proposes to continue to operate Unit 1 and Unit 2 at the MUR uprated power for up to 72 hours subsequent to an LEFM system becoming non-functional. In accordance with the proposed TRM, if the LEFM system is declared non-functional (i.e., "Alert" or "Fail" condition), the Technical Requirement (TR) will require that either the LEFM system be restored to functional status within 72 hours or power is to be reduced to ≤ 2775 MWt.

The electronics cabinet performs continuous monitoring of LEFM CheckPlus system parameters to identify any problems with the instrumentation. The LEFM self-verification feature provides a

comprehensive check of electronics, timing, signal-to-noise ratio, signal amplitude, noise levels, and average non-fluid delay. These features are described in detail in the LEFM topical reports.

An LEFM CheckPlus system Alert alarm indicates a loss of redundancy, and the calculated power level error associated with the LEFM CheckPlus flow measuring system in this condition is increased.

An Alert alarm is caused by:

- Loss of a single process input
- Loss of a single flow plane (loss of one or more flow transducers in a flow plane) on one or more feedwater lines
- Process input or output is calculated outside a pre-determined allowable range by one processing unit
- Internal self-check indicates system parameters that exceed pre-established limits and affect a single plane

An LEFM CheckPlus system Fail alarm indicates a loss of function. A Fail alarm is caused by:

- Loss of both redundant process inputs
- Loss of both flow planes in any individual feedwater meter
- Loss of both redundant spool piece RTDs on a single loop
- Loss of both feedwater header pressure inputs
- Failure of both redundant components in the electronics unit
- A process input or output is calculated outside a pre-determined allowable range by both PLCs
- Internal self-check indicates system parameters that exceed pre-established limits and affect multiple planes in any loop

In the event the LEFM CheckPlus system status changes to either Alert or Fail, Operations personnel are alerted through an annunciator in the main control room (CR). The PPC will also provide a computer alarm message to the CR if the status of the LEFM instrumentation changes.

The basis for the proposed 72-hour allowed outage time (AOT) is as follows:

- A completion time of 72 hours provides plant personnel sufficient time to plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.
- During the AOT, when the LEFM system is non-functional, the "normalized" feedwater flow from the venturis will be used for the calorimetric until the LEFM is returned to functional status. To ensure that the venturi based calorimetric is consistent with the LEFM CheckPlus system based calorimetric, the venturi-based flow rate is corrected to the most recent good value provided by the LEFM measurements as described in Section I.1.E.

- A review of flow venturi fouling history demonstrates that fouling/de-fouling should not introduce significant error/drift over a 72-hour period. This indicates that, without application of a bias based upon a bounding value of RTP secondary calorimetric uncertainty, Farley Unit 1 and Unit 2 can be operated for 72 hours without exceeding the licensed RTP limit when the flow venturi signals are used as an input to the Secondary Calorimetric portion of the RTP calculation in place of the LEFM system.
- As described in Cameron Report ER-157P (Reference I.2), the LEFM CheckPlus consists of two redundant planes of transducers and a single path or single plane malfunction results in a minimal increase in feedwater flow uncertainty. For Farley, operators will conservatively respond to a single path or single plane failure in the same manner as a complete system failure. This approach will simplify operator response and prevent misdiagnosing a failure mode.
- Operators routinely monitor other indications of core thermal power, including nuclear instrumentation system power range monitors, loop Δ-temperatures, steam flow, feed flow, turbine first stage pressure, and main generator (MG) output.

A CR annunciator response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine the LEFM CheckPlus system status and cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operator and maintenance response actions.

The limitations discussed above regarding operation with a non-functional LEFM CheckPlus system will be included in the TRM and associated implementation procedures, which will be revised prior to implementation.

As long as the LEFM system is functional, reactor power will be calculated utilizing the LEFM flow. If the LEFM system becomes non-functional, reactor power will be calculated utilizing the venturi feedwater flow normalized to the last good value provided by the LEFM feedwater flow. If at the end of the Completion Time, the LEFM system is not functional, reactor power will be calculated based on venturi feedwater flow assuming a 2-percent uncertainty and reactor power will be reduced to pre-MUR reactor power limitations.

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:

As described previously, these actions are covered in the proposed TRM. The LEFM technical limiting condition for operation requires that if an LEFM system is declared as non-functional and is not restored to functional status within 72 hours, then power is to be reduced to ≤ 2775 MWt.

References for Section I:

I.1 Engineering Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," March 1997.

I.2 Engineering Report ER-157(P-A), Revision 0, Revision 8, and Revision 8 Errata, "Supplement to Cameron Topical Report ER-BOP, "Basis for Power Uprates with an LEFM Check or a CheckPlus System," May 2008.

I.3 NRC letter from John N. Hannon to C. Lance Terry, TU Electric, "Comanche Peak Steam Electric Station Units 1 and 2 - Review of Caldon Engineering Topical Report ER-80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System' (TACS Nos. MA2298 and MA2299)," March 8, 1999.

I.4 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 (ML 102160663)

I.5 Cameron Caldon® Ultrasonics Engineering Report 1180P, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM✓+ System," May 2019.

I.6 Cameron Caldon® Ultrasonics Engineering Report 1181P, Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM✓+ System," May 2019.

I.7 Cameron Caldon® Ultrasonics Engineering Report 1182P, Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 1," July 2019.

I.8 Cameron Caldon® Ultrasonics Engineering Report 1183P, Rev. 1, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 2," July 2019.

I.9 WCAP-12771-P, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for Alabama Power Farley Nuclear Plant Units 1 and 2," December 2011

I.10 ASME NQA-1a-1999, Quality Assurance Requirement for Nuclear Facility Application, Subpart 2.7

I.11 Cameron Caldon® Ultrasonics Engineering Report 764, Rev. 0, "The Effect on the Distribution of the Uncertainty in Steam Moisture Content on the Total Uncertainty in Thermal Power," September 2009

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 <u>The requested uprate in power level continues to be bounded by the existing analyses of record for the plant</u>

II.1.B.ii <u>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</u>

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:

A review of Final Safety Analysis Report (FSAR) Chapter 15 was performed to support the Farley Units 1 and 2 Measurement Uncertainty Recapture Power Uprate (MUR-PU) with respect to the accident analyses. The FSAR review was conducted to confirm that the existing analyses of record, as currently presented in the FSAR, were either performed conservatively and remain valid and bounding for the proposed power uprate or were explicitly reanalyzed.

Table II.1-1 provides a brief overview of the accident/transient analyses and other analyses contained in the Farley Units 1 and 2 FSAR, the assumed core power level in each analysis, and whether these analyses remain bounding for the MUR-PU. This table also provides references to the NRC's previous approval of each analysis, if applicable, or a statement confirming that NRC approved methods were used in the analysis of record (AOR) that was implemented under the provisions of 10 CFR 50.59.

The analyses generally address the core and/or nuclear steam supply system (NSSS) thermal power in one of three ways and were correspondingly evaluated for MUR-PU conditions as follows:

1. Analyses That Apply a 2.0-Percent Increase to the Initial Power Level to Account for the Power Measurement Uncertainty

   These analyses would normally not have to be re-performed to address MUR-PU conditions because the sum of the proposed core power level increase and the decreased power measurement uncertainty falls within the previously analyzed conditions.

   For the FSAR Chapter 15 non-loss of coolant accident (LOCA) transient analyses in this category, a further explanation of the power levels utilized in the analyses follows:

   The existing non-LOCA licensing basis analyses (LBAs) support a nominal core power of 2775 MWt and a nominal NSSS power of 2785 MWt. For events analyzed using the standard thermal design procedure (STDP), a 2.0-percent increase in the initial NSSS power level was previously applied to account for the power measurement uncertainty. As such, the existing LBA STDP analyses support a maximum NSSS power level of 2840.7 MWt (1.02 x 2785 MWt). For the MUR-PU, the previous 2.0-percent uncertainty is reallocated so that a portion is applied to an uprated core power, and the remainder is retained to accommodate the revised (reduced) power measurement uncertainty.

   The events that required a reanalysis for the MUR-PU and PAD5 implementation were performed at a conservatively high NSSS power of 2841 MWt, which covers any combination of nominal uprated power level plus calorimetric uncertainty. However, based on the STDP event analyses previously described, the overall maximum NSSS power level (including power measurement uncertainty) supported by the MUR-PU program remains 2840.7 MWt.

   During the evaluation process for the MUR-PU, several legacy issues associated with the AOR for the Uncontrolled Boron Dilution (UBD) (FSAR Section 15.2.4) analysis were identified. The analysis was re-performed to address these legacy issues and take into consideration applicable adjustments to the inputs based on MUR-PU conditions. For the purposes of this submittal, this analysis has been included in Section III.1 for accidents/transients that are not considered to be bounded. An additional level of detail has been included to summarize the salient information from the re-analysis.

2. Analyses That Are Performed at 0-Percent Power Conditions

   These analyses would normally not have to be re-performed to address MUR-PU conditions because they are not dependent on power. However, as discussed in Section III.1, zero-power analyses cases from FSAR Sections 15.2.10 and 15.4.6 were reanalyzed for the purposes of this submittal. These analyses have been included in Section III.1 for accidents/transients that are not considered to be bounded, and an additional level of detail has been included to summarize the salient information from the analysis.

3. Analyses That Employ a Nominal Power Level

   These analyses have been re-performed for the proposed MUR-PU power level.

   In the LBA for events using the revised thermal design procedure (RTDP), an explicit initial condition uncertainty is not applied to NSSS power. For the RTDP methodology, uncertainties in plant initial conditions and other factors are statistically combined and accounted for in the departure from nucleate boiling ratio (DNBR) limit value. Therefore,

RTDP analyses must be addressed for the MUR-PU, since the nominal core power is increased.

RTDP analyses are performed in two ways: 1) transient statepoints are generated in the transient system code for use in a separate detailed DNBR calculation, or 2) DNBR is calculated directly by the transient system code(s). The transient statepoints include nuclear power and/or core heat flux as a fraction of nominal (FON) value. For analyses that generate transient statepoints, these statepoints were confirmed to remain applicable at MUR-PU conditions; the increased core power was applied to the statepoints in the DNBR calculation. For these statepoint analyses, a MUR-PU core power of 2823 MWt was analyzed. This represents an increase of approximately 1.7 percent over the current core power level (1.017 x 2775 MWt), anticipating that the revised power measurement uncertainty will be approximately 0.3 percent or greater. For RTDP analyses in which DNBR is calculated directly by the transient system code, a conservatively high core power of 2831 MWt (with corresponding NSSS power of 2841 MWt) was modeled.

II.1.D.i Reactor Trip System/Engineered Safeguards Features Actuation System Allowable Values

RESPONSE:

The safety analyses performed for the MUR-PU did not adjust the Reactor Trip System (RTS) or Engineered Safety Features Actuation System (ESFAS) nominal setpoints or allowable values from the non-uprated values. Therefore, the setpoints and allowable values remain unchanged from those documented in Technical Specification (TS) Tables 3.3.1-1 and 3.3.2-1 (Reference II.2).

The SG level control and protection uncertainties were evaluated based upon changes for the MUR-PU program. The SG level control and protection uncertainties experienced negligible changes because of the MUR-PU. All other RTS / ESFAS functions and safety analysis limits (SALs) are unaffected by the MUR-PU.

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

RESPONSE:

Departure from nucleate boiling (DNB) related analyses are in FSAR Sections 15.2.1, 15.2.3, 15.2.11, 15.2.14, 15.3.2, and 15.4.2. DNB-related analyses described in FSAR Sections 15.2.2, 15.2.5, 15.2.7, 15.2.8, 15.2.9, 15.2.10, 15.2.12, 15.3.4, and 15.4.4 were reanalyzed as described in Section III.1.

II.1.D.iii Discussion of RIS 2002-03 Section 11.1 Events

RESPONSE:

1. Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition – FSAR Section 15.2.1

   The AOR for rod withdrawal from subcritical (RWFS) is performed to demonstrate that the DNB design basis is met and the peak fuel pellet centerline temperature is less than the minimum temperature that could cause fuel melting.

The initial conditions for this event are not affected by the MUR-PU, because the event is analyzed at 0 percent power. The AOR statepoints, which include power level as FON values, are unaffected by the increased power level, because the time of reactor trip on the power-range neutron flux – low setpoint is also based on a FON condition (35 percent). Therefore, the time of trip is negligibly impacted.

The RWFS event was evaluated to support the implementation of the PAD5 fuel performance analysis and design model (Reference II.3), which affects the RWFS analysis inputs of fuel thermal conductivity and fuel melting temperature limit, both of which vary as a function of fuel burnup. The RWFS evaluation considered bounding thermal conductivity inputs at 0 MWD/MTU and 65,000 MWD/MTU burnup for the DNB and fuel pellet centerline temperature cases, respectively. With respect to the peak fuel pellet centerline temperature, the thermal conductivity is more limiting with PAD5, but the PAD5 fuel melting temperature limit is less limiting. The evaluation concluded that the AOR DNB statepoint remains bounding. The AOR statepoint was evaluated with the increased nominal core heat flux associated with the MUR-PU, and it was confirmed that the DNB design basis is satisfied. The evaluation also determined that the peak fuel pellet centerline temperature is less than the AOR value, due to the use of excessively conservative inputs in the AOR.

Therefore, the results and conclusions presented in FSAR Section 15.2.1 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

2. Uncontrolled RCCA Bank Withdrawal at Power – FSAR Section 15.2.2

   See Section III.1.

3. RCCA Misalignment – FSAR Section 15.2.3

   As described in the FSAR, the rod control cluster assembly (RCCA) misalignment events include one or more dropped RCCAs within the same group, a dropped RCCA bank, and statically misaligned RCCAs. The analyses of these events are performed to demonstrate that the DNB design basis is met, and the peak fuel pellet centerline temperature is less than the minimum temperature that could cause fuel melting. A cycle-specific calculation is performed for each reload cycle to ensure that the statically misaligned RCCA criterion will be satisfied for the specific operating conditions of that cycle.

   The AOR statepoints for the dropped RCCA(s) / bank transients are based on generic 3-loop plant analyses, which are further penalized to account for changes associated with the rod control system optimization for the Farley units. The statepoints are applied as changes from initial conditions, which makes them relatively independent of the core thermal power, such that the small change in core thermal power due to the MUR-PU does not impact the use of the generic statepoints for the dropped rod analysis. Additionally, the dropped rod analysis uses the minimum fuel rod temperatures, and therefore the transient statepoints are not adversely affected by PAD5 implementation.

   The AOR RCCA misalignment transients (i.e., dropped RCCAs, dropped RCCA bank, and statically misaligned RCCAs) were evaluated for the increased nominal core heat flux, and it was confirmed that the DNB design basis continues to be met. The evaluation also

confirmed that the peak fuel pellet centerline temperature is less than the minimum temperature that could cause fuel melting.

Therefore, the results and conclusions presented in FSAR Section 15.2.3 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

4. Uncontrolled Boron Dilution – FSAR Section 15.2.4

   See Section III.1.

5. Partial Loss of Forced Reactor Coolant Flow – FSAR Section 15.2.5

   See Section III.1.

6. Startup of an Inactive Reactor Coolant Loop – FSAR Section 15.2.6

   As described in FSAR Section 15.2.6, if a unit were to operate with one reactor coolant pump (RCP) out of service, there would be reverse flow through the inactive loop due to the pressure difference across the reactor vessel (RV). The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power with an inactive loop, and assuming the secondary side of the SG in the inactive loop is not isolated, there is a temperature drop across the SG in the inactive loop. With the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature. Starting an idle RCP without first bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity excursion and subsequent core power increase due to the moderator density reactivity feedback effect.

   Such an increase in core power could lead to DNB in the core. The startup of an inactive reactor coolant loop (RCL) event was evaluated with respect to the MUR-PU and PAD5 implementation.

   As Farley TSs Limiting Condition for Operation (LCO) 3.4.4 requires all three reactor coolant system (RCS) loops to be in operation during Modes 1 and 2, the maximum initial core power level for the startup of an inactive RCL event is approximately 0 MWt. Under these conditions, there can be no significant reactivity insertion because the RCS is initially at a nearly uniform temperature. Thus, there will be no increase in core power, and no automatic or manual protective action is required. A startup of an inactive RCL event would consist of an increase in the reactor coolant flow while the core remains in a shutdown or zero power condition. No analysis of the event is required.

   The MUR-PU and PAD5 implementation will not affect TS LCO 3.4.4, which requires all three RCS loops to be in operation during Modes 1 and 2. As such, it remains that no analysis of the startup of an inactive RCL event is required. The evaluation of the event concluded that the MUR-PU and PAD5 implementation are acceptable with respect to the startup of an inactive RCL.

   Therefore, the results and conclusions presented in FSAR Section 15.2.6 remain valid at MUR-PU conditions. The conclusion that no analysis of this event is required is unaffected by the MUR-PU.

7. Loss of External Electrical Load and/or Turbine Trip – FSAR Section 15.2.7

   See Section III.1.

8. Loss of Normal Feedwater – FSAR Section 15.2.8

   See Section III.1.

9. Loss of All AC Power to the Station Auxiliaries – FSAR Section 15.2.9

   See Section III.1.

10. Excessive Heat Removal Due to Feedwater System Malfunctions – FSAR Section 15.2.10

    See Section III.1.

11. Excessive Load Increase Incident – FSAR Section 15.2.11

    As described in the FSAR, the excessive increase in secondary system steam flow, or excessive load increase (ELI) incident, is analyzed to demonstrate that the DNB design basis is met. The ELI analysis is performed with the minimum fuel rod temperatures, and therefore the AOR is not adversely affected by PAD5 implementation.

    The ELI AOR was evaluated with respect to the increased core power (2831 MWt) for the MUR-PU. It was concluded that the DNB design basis continues to be met.

    Therefore, the results and conclusions presented in FSAR Section 15.2.11 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

12. Accidental Depressurization of the RCS – FSAR Section 15.2.12

    See Section III.1.

13. Accidental Depressurization of the Main Steam System – FSAR Section 15.2.13

    Due to the size of the break and the assumed initial conditions, an accidental depressurization of the main steam system (MSS) event is bounded by the main steam line rupture accident addressed in FSAR Section 15.4.2.1, "Rupture of a Main Steam Line."

    As such, no explicit analysis is performed for the accidental depressurization of the MSS. All applicable acceptance criteria are shown to be met via the conclusions in FSAR Section 15.4.2.1. Therefore, the results and conclusions presented in FSAR Section 15.2.13 remain valid at MUR-PU conditions. The conclusion, that no analysis of this event is required, is unaffected by the MUR-PU.

14. Inadvertent Operation of ECCS During Power Operation – FSAR Section 15.2.14

    As described in FSAR Section 15.2.14, the inadvertent operation of the emergency core cooling system (IOECCS) during power operation event could be caused by operator error, test sequence error, or a false electrical actuation signal. Following the actuation signal, the suction of the coolant charging pumps diverts from the volume control tank (VCT) to the refueling water storage tank (RWST). Simultaneously, the valves isolating the charging pumps from the injection header automatically open and the normal charging line isolation valves close. The charging pumps force the borated water from the RWST through the pump discharge header, the injection line, and into the cold leg of each loop. The passive

accumulator tank safety injection (SI) and low-head system are available, but they do not provide flow when the RCS is at normal pressure.

An SI signal normally results in a direct reactor trip and a turbine trip (TT). However, any single fault that actuates the emergency core cooling system (ECCS) will not necessarily produce a reactor trip. With no reactor trip, the reactor undergoes a negative reactivity excursion due to the injected boron, which causes a decrease in reactor power. The power mismatch causes a drop in the RV average temperature (T_{avg}) and consequent coolant shrinkage. The pressurizer pressure and water level decrease. Load decreases due to the effect of reduced steam pressure on load after the turbine throttle valve is fully open. If automatic rod control is used, these effects will lessen until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system (RPS) low pressurizer pressure trip or by manual trip.

Two IOECCS event cases are analyzed for Farley. One case addresses concerns related to DNB in the core, and the other case addresses concerns related to pressurizer filling. Each case was evaluated with respect to the MUR-PU and PAD5 implementation.

DNB Case Evaluation

In the Farley AOR for the IOECCS DNB case, nominal initial conditions are applied in the transient analysis. Following initiation of an IOECCS event, cold, borated water is injected into the RCS, causing the core power and RCS average temperature and pressure to rapidly decrease until the reactor is tripped due to low pressurizer pressure. Although the decreasing pressure has an adverse impact on the DNBR, this effect is outweighed by the DNBR benefits of the decreasing core power and RCS average temperature, so the minimum DNBR is the DNBR at event initiation. Therefore, the IOECCS event is not a concern relative to DNB. This does not change with the MUR-PU and PAD5 implementation, although the initial DNBR value will be slightly lower because of the increased initial power level. Consequently, no explicit analysis of the DNB case was performed for the MUR-PU and PAD5 implementation

Pressurizer Filling Case Evaluation

In the Farley AOR for the IOECCS pressurizer filling case, the initial NSSS power level is modeled as 102 percent of 2785 MWt, which is 2840.7 MWt.

Regarding PAD5 implementation, it has been determined qualitatively that the AOR for the IOECCS pressurizer filling case remains valid for the following reasons. The fuel temperatures are not critical parameters for the IOECCS analysis. Sensitivity studies have shown PAD5 fuel temperatures to have a relatively insignificant impact on IOECCS analysis results. The overall conservatism applied in the treatment of the initial core stored energy in IOECCS analyses would offset the increase in maximum fuel rod temperatures associated with PAD5 implementation.

The IOECCS transient analysis was evaluated for the MUR-PU and PAD5 implementation. For the IOECCS DNB case, the MUR-PU and PAD5 implementation will not change the fact that the minimum DNBR value will occur at event initiation, and therefore the IOECCS event is not a concern relative to DNB. For the IOECCS pressurizer filling case, relative to the MUR-PU, the analyzed power level is sufficient to cover an uprated NSSS power level

(including uncertainty) of up to 2840.7 MWt. For the IOECCS pressurizer filling case, relative to PAD5 implementation, the inputs that would be affected by PAD5 fuel temperatures are not critical parameters. In addition, the impact of PAD5 fuel temperatures is expected to be minor, and the conservatism applied in the treatment of the initial core stored energy would offset the minor impact of the PAD5 fuel temperatures.

In conclusion, the MUR-PU and PAD5 implementation are acceptable with respect to the IOECCS transient analysis because the results and conclusions presented in FSAR Section 15.2.14 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

15. Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuate Emergency Core Cooling System – FSAR Section 15.3.1

    The small break loss of coolant accident (SBLOCA) AOR for Farley Units 1 and 2 uses 10 CFR Part 50, Appendix K methodology with an RTP of 2775 MWt. The licensing basis methodology applies a 2 percent calorimetric power measurement uncertainty to the RTP, for an analyzed core power of 2831 MWt, in accordance with the original requirements of 10 CFR 50, Appendix K. The inputs applied in the analyses were confirmed to remain applicable, bounding, or negligibly changed under MUR-PU conditions. Based on the existing 2 percent power measurement uncertainty margin included in the SBLOCA analysis, a MUR-PU within 2 percent is bounded by the current Farley Units 1 and 2 SBLOCA AOR.

    The reported 10 CFR 50.46 results in FSAR Section15.3.1 remain applicable and meet the acceptance criteria of 10 CFR 50.46. Therefore, the results and conclusions presented in FSAR Section 15.3.1 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

16. Minor Secondary System Pipe Breaks – FSAR Section 15.3.2

    Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Because the results of analysis presented in Subsection 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analysis for minor secondary system pipe breaks is not required.

    The analysis presented in FSAR Section 15.4.2 demonstrates that the consequences of a minor secondary system pipe break are acceptable because a DNBR of less than the limit value does not occur even for a more critical major secondary system pipe break. Therefore, the results and conclusions presented in FSAR Section 15.4.2 remain valid at MUR-PU conditions. The conclusion, that no analysis of this event is required, is unaffected by the MUR-PU.

17. Inadvertent Loading of a Fuel Assembly Into an Improper Position – FSAR Section 15.3.3

    Fuel and core loading errors can arise from the inadvertent loading of one or more fuel assemblies into improper positions, the loading of a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. These loading errors will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of

lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

Fuel assembly loading errors are prevented by administrative procedures and detected by monitoring of the core power distribution with in-core instrumentation. Operation at MUR-PU conditions and PAD5 implementation does not affect the ability of the in-core instrumentation to detect the inadvertent loading and subsequent operation with a fuel assembly in an improper position.

The results and conclusions presented in FSAR Section 15.3.3 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

18. Complete Loss of Forced Reactor Coolant Flow – FSAR Section 15.3.4

    See Section III.1.

19. Waste Gas Decay Tank Rupture – FSAR Section 15.3.5

    The waste gas decay tank rupture accident is defined as an unexpected and uncontrolled release of radioactive xenon and krypton fission product gases stored in a waste decay tank as a consequence of a failure of a single gas decay tank or associated piping. The gaseous waste processing system (GWPS) is designed to remove fission product gases from the reactor coolant and has the capacity to contain these gases throughout the plant life. At MUR-PU conditions, the required containment, confinement, and filtering capacities of the GWPS and the capacities of its various decay and storage tanks are sufficient because the MUR-PU does not materially affect the system flow rates or gas volumes.

    In accordance with current licensing basis as approved by the NRC in Reference II.32, the dose consequences of the waste gas tank rupture must remain within 500 mrem whole body.

    The isotopic gas inventory used in the current waste gas decay tank rupture analysis is based on a coolant inventory reflective of a core thermal power of 2831 MWt (i.e., 102 percent of the current RTP of 2775 MWt), which bounds operation at MUR-PU operating conditions. Therefore, the results and conclusions presented in FSAR Section 15.3.5 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by NRC per the references listed in Table II.1.

20. Single RCCA Withdrawal at Full Power – FSAR Section 15.3.6

    No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The event analyzed must result from multiple wiring failures, multiple significant operator errors, or subsequent and repeated operator disregard of event indication. The probability of such a combination of conditions is low so that the limiting consequences may include slight fuel damage.

    The single RCCA withdrawal at full power (single rod withdrawal at power (RWAP)) incident is analyzed to demonstrate that no more than 5 percent of the total number of fuel rods in the core would experience a DNB. The analysis discussion presented in the FSAR was

evaluated with respect to the increased core power for the MUR-PU and PAD5 implementation. It was concluded that less than 5 percent of the total number of fuel rods in the core would experience DNB. A cycle-specific calculation is performed for each reload cycle to ensure that the Single RCCA Withdrawal at Full Power rods-in-DNB limit is met.

The results and conclusions presented in FSAR Section 15.3.6 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

21. Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accidents) – FSAR Section 15.4.1

FSAR Section 15.4.1 describes the current best estimate large break loss of coolant accident (BE LBLOCA) analysis performed for Farley Units 1 and 2. The analysis used the best estimate Automated Statistical Treatment of Uncertainty Method (ASTRUM) methodology as approved in Reference II.24 for calculation of peak cladding temperature (PCT) and oxidation (local and core-wide). The BE LBLOCA analysis is performed at an assumed core power of 102 percent of 2775 MWt (2831 MWt), which bounds operation considering MUR-PU conditions.

The reported 10 CFR 50.46 results in FSAR Table 15.4-3, including all additional evaluations as described in FSAR Section 15.4.1.5.3, remain applicable and meet the acceptance criteria of 10 CFR 50.46.

Post- LOCA long-term core cooling (LTCC) analyses use core power in the calculation of core boiloff in the sump recirculation phase after a LOCA. The current post-LOCA LTCC analyses for Farley Units 1 and 2 assume a nominal core power of 2,775 MWt plus an additional 2-percent calorimetric power measurement uncertainty (yielding an assumed core power of 2,831 MWt). Consistent with the requirements contained in Appendix K of 10 CFR 50, the decay heat assumed in the LOCA LTCC analysis is 1.2 times the values for infinite operating time in the 1971 ANS Standard. The total core power (2831 MWt) assumption used in core boiloff calculations in the LTCC analyses is consistent with the core power reflected in the MUR-PU NSSS design parameters. Therefore, there is no impact on the Farley Units 1 and 2 LTCC analyses due to the MUR-PU.

In accordance with current licensing basis which incorporates the full implementation of AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following a LOCA meet the onsite and offsite dose limits set by 10 CFR 50.67, as modified by Regulatory Guide (RG) 1.183, Revision 0. The inventory of radionuclides in the reactor core available for release into containment following a LOCA is currently based on a core thermal power of 2831 MWt (102 percent of the current RTP of 2775 MWt), which bounds operation at MUR-PU operating conditions. Additional factors that can affect the equilibrium core inventory and therefore environmental releases are fuel enrichment and burn-up, both of which remain unchanged by the MUR-PU. The current masses, volumes and boron concentrations of the reactor coolant system, accumulators, and RWST as specified by the plant TSs and TRM remain unchanged. Current licensing basis LOCA mass and energy (M&E) releases (and, consequently, the post-LOCA containment environmental conditions), are not impacted by the MUR-PU. This is relevant because the post-LOCA containment transients (such as containment pressure, sump water temperatures, etc.) have

the potential of impacting fission product removal coefficients / iodine re-evolution, etc. Therefore, the MUR-PU will have no significant effect on the LOCA dose consequences reported in FSAR Chapter 15.

Therefore, the results and conclusions presented in FSAR Section 15.4.1 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

22. Major Secondary System Pipe Rupture – FSAR Section 15.4.2

15.4.2.1 Rupture of Main Steam Line

The AOR for main steam line break (MSLB) are performed to demonstrate that the DNB design basis is met, and the peak fuel pellet centerline temperature is less than the minimum temperature that could cause fuel melting.

Hot Zero Power Steam Line Break

The hot zero power (HZP) SLB event is analyzed at zero power initial conditions and therefore the transient analysis is not impacted by the MUR-PU. The analysis is performed with minimum fuel rod temperatures, and therefore the AOR statepoints are not adversely affected by PAD5 implementation.

The HZP SLB analysis was evaluated using the AOR statepoints and the increased nominal core heat flux. It was confirmed that the DNB design basis continues to be met. The evaluation also determined that the peak linear heat rate (kW/ft) does not exceed a value corresponding to the minimum fuel centerline temperature that could cause fuel melting, which is a function of fuel burnup with the PAD5 implementation.

Hot Full Power Steam Line Break

The hot full power (HFP) SLB event is analyzed at 100-percent power initial conditions to demonstrate that core protection is maintained prior to and immediately following reactor trip.

An evaluation determined that the transient response for the HFP SLB from a higher initial power corresponding to the MUR-PU does not substantially affect the transient statepoints for the event, such that the limiting transient statepoints from the AOR remain valid. Furthermore, it was confirmed that the fuel rod temperatures used in the transient analysis do not adversely affect the results, and thus the AOR transient statepoints are not affected by PAD5 implementation.

The HFP SLB analysis was evaluated using the AOR statepoints and the increased nominal core heat flux. It was confirmed that the DNB design basis continues to be met. The evaluation also determined that the peak linear heat rate (kW/ft) does not exceed a value corresponding to the minimum fuel centerline temperature that could cause fuel melting, which is a function of fuel burnup with the PAD5 implementation.

The steam releases for radiological dose calculation assumes a fuel temperature value that is used to determine the stored energy in the core. The steam releases for dose calculation assumes the stored energy in the core as 6.54 full-power seconds (fps) based on a generic maximum fuel temperature that is conservatively high. The calculated value for core stored energy for Farley Units 1 and 2 at the MUR-PU power after accounting for the effects of the

PAD5 model is 3.56 fps. The 6.54 fps used in the calculation is highly conservative when compared to the current calculated Farley core stored energy. Therefore, the steam releases calculation remains valid and bounding for use as input to the subsequent dose consequence analyses. The steam releases have been calculated at the uprated NSSS power level of 2841 MWt for the SLB event and were provided as input to the radiological dose analyses to support the Farley Units 1 and 2 MUR-PU.

In accordance with current licensing basis which incorporates the full implementation of AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following the SLB meet the onsite and offsite dose limits in 10 CFR 50.67 and the guidance in RG 1.183, Revision 0.

A comparison of the secondary system mass flows used in the current dose consequence analysis for SLB to the associated MUR-PU values, indicates that the secondary system mass flows used in the analysis include sufficient margin to bound the values applicable to MUR- PU operation. Therefore, the MUR-PU will have no effect on the SLB dose consequences reported in FSAR Chapter 15.

In conclusion, the results and conclusions presented in FSAR Section 15.4.2.1 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

See Section III.1 for FSAR Section 15.4.2.2, "Major Rupture of a Main Feedwater Pipe."

23. Steam Generator Tube Rupture – FSAR Section 15.4.3

The steam generator tube rupture (SGTR) accident analysis demonstrates that the radiological consequences are less than the regulatory limits as discussed in Reference II.19. The primary to secondary mass release and the amount of steam vented from each of the SGs used in the dose analysis was obtained from M&E balance calculations.

The MUR-PU would tend to increase the vessel outlet temperature and reduce the SG outlet pressure, which could adversely affect the SGTR break flow and steam release calculations and the calculation of SGTR break flow flashing fractions. However, plant operational limits with the MUR-PU will limit the vessel outlet temperature to ≤ 613.3°F, and the full load steam pressure to ≥ 690 psia. With these limitations in place, the SGTR calculations remain bounding.

The increase in power level and main feedwater temperature associated with the MUR-PU would tend to reduce the initial secondary side water mass. Within the methodology of the M&E balance calculations, a higher initial SG secondary side water mass is conservative and results in higher post-trip steam releases. Therefore, the impact of the MUR-PU on the initial secondary side water mass would not result in increased releases and does not adversely affect the SGTR input to dose.

The SGTR analysis modeled a core power of 2775 MWt and did not include a 2 percent power uncertainty, however an approximate 10 percent increase in the post-trip steam releases was included in the radiological consequences evaluation and this 10 percent increase more than offsets the 2 percent power increase. The pre-trip steam releases to the condenser are not calculated as part of the SGTR analysis, so the nominal steam flow rate was provided for use in the dose analysis. Therefore, the only adverse effect of the MUR-PU

is that the pre-trip steam release rate used in the dose analysis would be increased. To conservatively bound the uprate, a higher value of 1.255E7 lbm/hr (1162.0 lbm/sec/SG) was considered in the dose evaluation. This is an increase of less than 3 percent over the current value of 1.224E7 lbm/hr (1133.3 lbm/sec/SG) for the pre-trip steam releases to the condenser from the ruptured and intact SGs. The increase in the pre-trip secondary system mass flow used for the SGTR from 1133.3 lbm/sec/SG to 1162.0 lbm/sec/SG following the MUR-PU will have a negligible impact on the dose consequences.

The SGTR analysis does not model a plant-specific or fuel-specific maximum fuel temperature but assumes a conservatively high fuel temperature which bounds the effects of PAD5. Thus, the SGTR analysis is not affected by PAD5 implementation.

Therefore, the results and conclusions presented in FSAR Section 15.4.3 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

24. Single Reactor Coolant Pump Locked Rotor – FSAR Section 15.4.4

    See Section III.1.

25. Fuel Handling Accident – FSAR Section 15.4.5

    The accident is defined as the dropping of a spent-fuel assembly onto the spent-fuel pool floor or the refueling canal floor resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations.

    In accordance with the current licensing basis that incorporates AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following Condition IV design basis accidents (DBAs), including the fuel handling accident, meet the onsite and offsite dose limits set by 10 CFR 50.67 and the guidance in R.G. 1.183, Revision 0.

    The inventory of radionuclides in the fuel gap of the dropped fuel assembly, which is released to the environment is currently based on a core thermal power of 2831 MWt (102 percent of the current RTP of 2775 MWt), which bounds operation at MUR-PU operating conditions. Additional factors that can affect the fuel gap inventory (and, therefore, environmental releases following a fuel handling accident) are fuel enrichment and burnup, both of which remain unchanged by the MUR-PU. In addition, the evaluation confirmed that the FΔH limit (maximum peaking factor) of 1.7 will not be challenged, and that the maximum linear heat generation rate will not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU at MUR-PU conditions. The above confirmed the continued applicability of the non-LOCA gap fractions presented in Table 3 of RG 1.183 for MUR-PU conditions. Therefore, the MUR-PU will have no significant effect on the fuel handling accident dose consequences reported in FSAR Chapter 15. The MUR-PU will have no significant effect on the dose consequences at the EAB, LPZ, and in the CR for DBAs reported in FSAR Chapter 15.

    Therefore, the results and conclusions presented in FSAR Section 15.4.5 remain valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1.

26. Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection) – FSAR Section 15.4.6

    See Section III.1.

27. Anticipated Transients Without Scram (ATWS) – FSAR Section 15.5

    See Section III.1.

28. Containment Analyses – FSAR Section 6.2

    FSAR Section 6.2.1.3.4.1, "LOCA Mass and Energy Releases"

    LOCA Long-Term Mass and Energy Releases

    The LBLOCA Long-Term M&E Releases analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical LOCA. The methodology for the most limiting LOCA M&E release calculation is contained in References II.28 and II.29. Based on this methodology, the AOR presently assumes a core thermal power of 2830.5 MWt. This is the licensed core power of 2775 MWt with an additional 2 percent calorimetric uncertainty applied to the 2775 MWt value to account for the power measurement uncertainty. The MUR-PU improved thermal power measurement accuracy obviates the need for the full 2 percent power margin assumed in the analysis.

    In addition, the analyzed loop average temperature does not change (583.2 °F), and the core stored energy used to initialize the core for the Farley Unit 1 and Unit 2 LOCA M&E release analysis is bounding when the effects of thermal conductivity degradation (TCD) have been included. Thus, the initial energy content of the RCS fluid does not change, and therefore the margin of safety would not be reduced. In summary, there is no effect on either the long-term LOCA M&E release AOR nor the associated conclusions in FSAR Section 6.

    Short-Term Loss of Coolant Accident Mass and Energy Release

    Short-term LOCA M&E release calculations are performed to support the reactor cavity and loop subcompartment (which includes the SG compartment and pressurizer compartment) pressurization analyses. These analyses are performed to guarantee that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a LOCA within the region.

    The short-term LOCA blowdown transients are characterized by the M&E releases that occur during a subcooled condition. The modified Zaloudek correlation, which models this condition, is currently used to calculate the short-term LOCA M&E releases (Reference II.30).

    The conclusions of the AOR remain applicable and bounding for the MUR-PU conditions.

    FSAR Section 6.2.1.3.10, "Containment Subcompartment Analyses"

    Subcompartment analyses are performed to guarantee that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse, which accompanies a high energy line rupture within the containment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the M&E release rates associated with the blowdown, the subcompartment volume, and vent paths. Because short-

term releases are linked directly to the critical mass flux, the short-term M&E release rates are primarily affected by the initial conditions in the RCS.

The evaluation of the impact of the MUR-PU on the short-term LOCA M&E releases concluded that the current M&E releases remain applicable for MUR-PU operations. Therefore, the reactor cavity wall, SG compartment wall, and pressurizer compartment wall analyses results documented in the FSAR remain applicable for the MUR-PU.

FSAR Section 6.2.1.3.11, "Main Steam Line Ruptures Inside Containment"

Long-Term Steam Line Break Mass and Energy Releases Inside Containment

The AOR applicable to Farley Units 1 and 2 for the inside containment long-term full-power steam line breaks (SLBs) assumes a 2 percent calorimetric uncertainty added to the NSSS power of 2,785 MWt. As long as the sum of the power increase and power calorimetric uncertainty does not exceed 2 percent, there is no effect on the current licensing-basis long-term SLB M&E release analysis.

The critical parameters for the long-term SLB event inside containment include the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the minimum plant shutdown margin, the initial value for the SG water mass, main feedwater flow, auxiliary feedwater (AFW) flow, main feedwater and AFW enthalpies, and the times at which steam line and feedwater line isolation occur. The safety analyses of the long-term SLB M&E releases inside containment do not explicitly input fuel temperatures but do model fuel-to-coolant heat transfer coefficients (UA) based on fuel temperatures that are determined by the PAD5 computer code. Fuel temperatures are not considered a key input, so any changes attributed to the PAD5 computer code, including TCD, will produce negligible to no impact on the transient M&E releases. The SLB M&E releases calculated or evaluated for the Farley Units 1 and 2 MUR-PU assume maximum UAs based on minimum fuel temperatures.

The minimum fuel temperatures resulting from the PAD5 calculations, relative to those used in the AOR, remain acceptable. Consequently, the AOR and the SLB M&E release remain valid and bounding even if the effects of the PAD5 computer code were explicitly modeled. The FSAR conclusions remain valid for the long-term SLB event inside containment, therefore, the transient M&E releases inside containment remain valid for use as input to the containment integrity analysis.

Containment Pressure and Temperature Analysis – FSAR Section 6.2

The MUR-PU has the potential to affect the design basis analyses that determine the containment temperature and pressure following a LOCA or other high energy line break (HELB). Each Farley containment building is designed to withstand an internal pressure of –3 psig to +54 psig and a temperature of 280 °F. The current AOR is performed using the GOTHIC 6.0 Code (Reference II.32). The containment temperature and pressure profiles are given in Section 6.2 of the FSAR.

The current M&E releases used to form the input for the AOR are based upon a core power level of 2831 MWt. The M&E releases for the AOR will not change for the MUR-PU. No change to the temperature and pressure profiles is expected to occur as a result of the MUR-PU. Therefore, the results and conclusions presented in FSAR Section 6.2 remain

valid at MUR-PU conditions. The methodology by which the AOR was performed was reviewed and approved by the NRC per the references listed in Table II.1-1.

29. Equipment Qualification – FSAR Section 3.11

The effect on the Farley nuclear equipment qualification analysis as a result of implementing the MUR-PU on the electrical equipment identified in the equipment qualification program for environmental qualification (EQ) was evaluated. This evaluation determined the following:

1. The equipment and components of the equipment qualification program will continue to operate satisfactorily and perform their intended functions at the uprated conditions to satisfy the requirements outlined in 10 CFR 50.49, and the safety-related electrical equipment is qualified to survive the environment at its specific location during normal operation and during an accident.
2. The equipment qualification program equipment will accommodate MUR-PU conditions without exceeding electrical equipment qualification margins for the parameters of temperature, pressure, radiation, and similar parameters, as defined by IEEE Standard 323-1974.

Based on a review of the conditions listed below, there are no changes that affect the EQ of the Farley equipment.

- Containment Pressure and Temperature Analyses
- Containment Flooding
- MSLB in the Main Steam Valve Room (MSVR)
- Other HELBs Pressure / Temperature Outside Containment
- Post- LOCA Sump Water pH
- Radiation Environments to Support Equipment Qualification

In conclusion, the existing AORs remain bounding, and the MUR-PU will not affect equipment in the equipment qualification program for EQ.

30. Flooding – FSAR Section 3.11

The Farley EQ program was implemented to comply with the requirements of NRC Inspection and Enforcement Bulletin 79-01B, NUREG-0588, Revision 1, and 10 CFR 50.49.

FSAR Section 3.11 provides information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features, the RPSs, and other safety-related systems are designed to guarantee acceptable performance during normal and DBA environmental conditions. FSAR Table 3.11-1 indicates that the post-accident submergence level of Elevation 115'-0" inside containment is used for EQ of components at Farley.

The MUR-PU will not affect the current masses of the RCS, accumulators, and the RWSTs. In addition, the long-term LOCA M&E releases remain applicable for MUR-PU operations, thus the containment sump water temperature transient is not affected by the MUR-PU.

Consequently, the current containment sump submergence level remains applicable for the MUR-PU.

31. Spent Fuel Pool Criticality and Loss of Cooling – FSAR Section 9.1.3

Inadvertent or accidental criticality in the new fuel pit (NFP) and spent fuel pool (SFP) will be prevented through compliance with the Farley Units 1 and 2 TSs, the geometric spacing of fuel assemblies in the NFP and SFP, and administrative controls imposed on fuel handling procedures. For Farley Units 1 and 2 the Design Features that preclude criticality are documented in TS 4.3.1 (Reference II.2). The Design Feature requirements are maintained through compliance with TS 3.7.14, and 3.7.15 (Reference II.2).

The NFP stores fuel before operation in the reactor, as such, the MUR-PU will not affect the fuel assemblies stored in the NFP. Additionally, there are no changes to fuel design planned for Farley Units 1 and 2. Therefore, the reactivity of the NFP will not change due to the MUR-PU, and the NFP will not be re-analyzed.

The SFP can store fuel both before and after operation in the reactor. The changes in reactor operation due to the MUR-PU can affect the reactivity of the SFP. Specifically, after the MUR-PU, fuel assemblies which are stored using burnup credit could have operating histories that are more limiting than fuel operated under pre-MUR-PU conditions. Therefore, the criticality AOR for the SFP, including the loss of cooling effect on criticality, was updated and submitted to the NRC under a separate license amendment request (SNC letter NL-19-0796).

Spent fuel pool cooling is addressed in Section VI.1.D of this Attachment.

32. Station Blackout – FSAR Section 5.5.2.3.2

Natural circulation cooldown is evaluated in response to the "upset condition" of a loss of alternating current (AC) power. The Farley Units 1 and 2 MUR-PU increases the decay heat produced by the reactor core relative to the current RTP. The MUR-PU does not impact the analysis for a natural circulation cooldown because the analysis was based on a power level of 2900 MWt which is higher than the MUR-PU power.

Therefore, the results and conclusions presented in FSAR Section 5.5.2.3.2 remain valid at MUR-PU conditions.

33. High Energy Line Breaks Outside Containment

<u>Main Steam Line Break Outside Containment Pressure and Temperature Analysis – FSAR Appendix 3J</u>

The critical parameters for the long-term SLB event outside containment include the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the minimum plant shutdown margin, the initial and trip values for the SG water mass, main feedwater flow, AFW flow, main feedwater and AFW enthalpies, and the times at which steam line and feedwater line isolation occur. The AOR applicable to Farley Units 1 and 2 for the outside containment long-term full-power SLBs assumes a 2 percent calorimetric uncertainty added to the NSSS power of 2785 MWt.

The safety analyses of the long-term SLB M&E releases outside containment do not explicitly input fuel temperatures but do model fuel-to-coolant heat transfer coefficients (UA)

based on fuel temperatures that are determined by the PAD computer code. Fuel temperatures are not considered a key input, so any changes attributed to the PAD5 computer code, including TCD, will produce negligible-to no impact on the transient M&E releases. The minimum fuel temperatures resulting from the PAD5 calculations, relative to those used in the AOR remain acceptable. The AOR and the SLB M&E release MUR-PU evaluations remain valid and bounding even if the effects of the PAD5 computer code are explicitly modeled. The conclusions remain valid for the long-term SLB event outside containment; therefore, this includes the MSVR temperature response outside containment. Explicitly modeling the effects of the PAD5 computer code related to the fuel average temperatures and core stored energy would not change the conservative reactivity feedback model used in the SLB M&E releases outside containment analysis. The long-term SLB M&E release analyses are not sensitive to the fuel-to-coolant heat transfer coefficient input that could change due to the fuel temperatures calculated using the PAD5 computer code. Therefore, the transient M&E releases outside containment remain valid for use as input to the outside containment MSVR temperature response analysis.

High-Energy Line Pipe Break (Outside Containment) - FSAR Appendix 3K

Safety related components located outside containment are designed to operate in the environmental conditions such as temperature, pressure, humidity and flooding resulting from a postulated HELB. With the exception of the main feedwater system and the portion of the AFWS from the junction with the main feedwater line to the first isolation valve, the current design basis system initial conditions remain unchanged or bound the MUR-PU system conditions.

The elevated feedwater / AFW temperature from 440°F to 446°F may increase the break effluent enthalpy and associated energy release. However, the areas in the auxiliary building affected by a rupture in the main feedwater system and the AFWS are the same as the MSS; therefore, the compartment pressurization and environmental consequences associated with the feedwater / AFW line breaks are limited to the MSVR and the pipe chase. Because the M&E release rates associated with a feedwater line break are lower than the M&E release rates for an MSLB, the pressures and temperatures in the MSVR and the pipe chase following a MSLB bound that associated with a feedwater line break. The minor estimated increase in the feedwater temperature is considered slightly beneficial to the flooding analysis due to the associated density effect and resulting reduction in the volumetric break flow rate following the main feedwater or AFW pipe rupture. Consequently, the results of the current flooding analysis documented in FSAR Section 3K remain bounding for the MUR-PU conditions.

The current MSVR pressure, temperature and flooding response to a postulated MSLB, along a spectrum of break sizes, remain valid for the MUR-PU. The current design basis compartment pressurization, flooding, and environmental effects response to postulated HELBs outside the containment remain valid for the MUR-PU.

34. Fuel Evaluations

Nuclear Design – FSAR Section 4.3

The standard set of reload core design criteria (Reference II.16) have been confirmed via evaluation or explicit analysis for up to a 2 percent increase in core thermal power (2831 MWt).

A review of all Reload Safety Analysis Checklist (RSAC) items for Farley Units 1 and 2 was performed to determine items that could potentially be challenged as a result of the MUR-PU. The fuel management strategy (feed batch size and peaking factor design limits) is not expected to change as a result of the small increase in core power.

It was confirmed that burnup dependent power-to-melt limits would be met for HZP SLB, HFP SLB, Condition II events and dropped rod analyses. The impact of the MUR-PU on all other RSAC parameters (peaking factors, reactivity coefficients, shutdown margin, control rod insertion limits, trip reactivity, boron dilution, control rod ejection, and SLB) was evaluated by comparison of typical margins to their limits. Adequate margin was confirmed to be available for both Farley Units 1 and 2 at MUR-PU conditions. Cycle-specific calculations are performed for each reload cycle. These cycle-specific analyses are performed to guarantee that all core design and RSAC criteria will be satisfied for the specific operating conditions of that cycle.

Fuel Rod Design – FSAR Section 4

See Section III.1

Core Thermal Hydraulic – FSAR Section 4

See Section III.1

Fuel Mechanical Design – FSAR Section 4.2

The fuel mechanical design analyses potentially affected by the MUR-PU include the fuel assembly lift force analysis, which specifically models the nominal core power, and the fuel seismic / LOCA analysis which relies upon seismic/LOCA core plate motions for evaluating the potential for grid crush. To support the MUR-PU conditions, the fuel assembly lift force analyses were performed at the MUR-PU core power. The resulting fuel assembly lift forces were then evaluated with respect to the top nozzle hold-down spring analyses. It was concluded that the applicable design criteria, including demonstrating that fuel assembly liftoff does not occur, are satisfied. The one exception is the turbine overspeed transient associated with a loss of load (LOL) event, which allows for fuel assembly liftoff. With respect to the fuel seismic/LOCA analyses, the current licensing basis seismic/LOCA core plate motions remain applicable for the MUR-PU. Therefore, the fuel assembly structural integrity is not affected for the seismic/LOCA event for the implementation of the MUR-PU. All of the remaining fuel assembly fuel mechanical design analyses that serve as the basis for the current 17x17 Vantage+ with debris mitigating features fuel design are not directly affected by the reactor core power level. Thus, the fuel mechanical design analyses remain valid and bounding for the MUR-PU. The results and conclusions presented in FSAR Section 4.2 remain valid at MUR-PU conditions.

References for Section II:

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II.2 Joseph M. Farley Nuclear Plant Technical Specifications, Unit 1 - Amendment 221, Unit 2 - Amendment 218, November 29, 2018.

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II.4 Westinghouse Report, WCAP-7979-A (Westinghouse Proprietary) and WCAP-8028-A (Nonproprietary),"TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," January 1975.

II.5 Westinghouse Report, WCAP-7908-A, "FACTRAN - A Fortran-IV Code for Thermal Transients in a UO2 Fuel Rod," December 1989. (Westinghouse Proprietary)

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II.11 Westinghouse Report, WCAP-10079-P-A, Revision 0, "NOTRUMP – A Nodal Transient Small Break and General Network Code," August 1985. (Westinghouse Proprietary)

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II.13 Westinghouse Report, WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (Westinghouse Proprietary)

II.14 Westinghouse Report, Appendices F and G to WCAP-12610, "LOCA NOTRUMP Evaluation Model: ZIRLO Modifications and Accident Evaluations, LOCA Plant Specific," December 1990. (Westinghouse Proprietary)

II.15 Westinghouse Report, Addendum 1-A to WCAP-12610-P-A and CENPD-404-P-A, "Optimized ZIRLO™," July 2006. (Westinghouse Proprietary)

II.16 Westinghouse Report, WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. (Westinghouse Proprietary)

II.17 Westinghouse Report, WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988. (Westinghouse Proprietary)

II.18 Westinghouse Report, WCAP- 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," September 1986. (Westinghouse Proprietary)

II.19 U.S. NRC Letter to SNC, "Joseph M. Farley Nuclear Plant, Units 1 and 2 – Issuance of Amendments Adopting Alternative Source Term, TSTF-448, Revision 3, and TSTF-312, Revision 1 (CAC Nos. MF8861, MF8862, MF8916, MF8917, MF8918, and MF8919; EPID Nos. L-2016-LLA-0017, L-2016-LLA-0018, and L-2016-LLA-0019)," December 2017. (ADAMS Accession Number ML17271A265).

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Mass and Energy Release Model for Containment Design - March 1979 Version' (TAC No. MC7980)," October 18, 2005.

II.30 Westinghouse Report, WCAP-8264-P-A, Revision 1, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," August 1975. (Westinghouse Proprietary). Westinghouse Report, WCAP-8312-A, Revision 2, "Topical Report Westinghouse Mass and Energy Release Data for Containment Design," August 1975. (Non-Proprietary)

II.31 "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Westinghouse Proprietary), WCAP-8860 (Nonproprietary), September 1976; "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture," WCAP-8822-S1-P-A (Westinghouse Proprietary), WCAP-8860-S1-A (Nonproprietary), September 1986; "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Sub-Atmospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary), WCAP-8860-S2-A (Nonproprietary), September 1986.

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Table II.1-1: FSAR Accidents, Transients, and Other Analyses

RIS 2002-03 Farley Units 1 and 2 FSAR Section	II.1.A Accident/Transient Title	II.1.B.i Power Level Used (MWt)	II.1.B.i Is Power Bounding for MUR? (Yes/No)	II.1.B.ii Approved by NRC or Conducted Using Methods / Processes Approved by the NRC	II.1.C Confirm That Bounding Event Determinations Remain Valid	II.1.D NRC Approval
(1) 15.2.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	0 MWt **(Note 2)**	Yes	References II.4, II.5	See section II.1	Analysis performed using NRC approved methodologies.
(2) 15.2.2	Uncontrolled RCCA Bank Withdrawal at Power	2840.7 MWt NSSS (overpressure) 2841 MWt NSSS (DNB)	No - See section III.1	References III.1, III.2, II.3, II.4	See section III.1	**Note 1**
(3) 15.2.3	RCCA Misalignment	2841 MWt NSSS **(Note 2)**	Yes	References II.6, II.7, II.8, II.9, II.16	See section II.1	Analysis performed using NRC approved methodologies.
(4) 15.2.4	Uncontrolled Boron Dilution	2841 MWt NSSS	Yes	Reference III.5	See section III.1	**Notes 1 and 3**
(5) 15.2.5	Partial Loss of Forced Reactor Coolant Flow	2841 MWt NSSS	No - See section III.1	References III.1, III.2, III.3, III.4	See section III.1	**Note 1**
(6) 15.2.6	Startup of an Inactive Reactor Coolant Loop	N/A	N/A	Precluded by Tech Specs	See section II.1	Precluded by Technical Specifications
(7) 15.2.7	Loss of External Electrical Load and/or Turbine Trip	2841 MWt NSSS	No - See section III.1	References III.4, III.6	See section III.1	**Note 1**
(8) 15.2.8	Loss of Normal Feedwater	2841 MWt NSSS	Yes	References III.4, III.6, III.7	See section III.1	**Note 1**

RIS 2002-03 Farley Units 1 and 2 FSAR Section	II.1.A Accident/Transient Title	II.1.B.i Power Level Used (MWt)	II.1.B.i Is Power Bounding for MUR? (Yes/No)	II.1.B.ii Approved by NRC or Conducted Using Methods / Processes Approved by the NRC	II.1.C Confirm That Bounding Event Determinations Remain Valid	II.1.D NRC Approval
(9) 15.2.9	Loss of All AC Power to the Station Auxiliaries	2841 MWt NSSS	Yes	References III.4, III.6, III.7	See section III.1	**Note 1**
(10) 15.2.10	Excessive Heat Removal Due to Feedwater System Malfunctions	2841 MWt NSSS	No - See section III.1	References III.1, III.2	See section III.1	**Note 1**
(11) 15.2.11	Excessive Load Increase Incident	2841 MWt NSSS	Yes	References II.8, II.10	See section II.1	Analysis performed using NRC approved methodologies.
(12) 15.2.12	Accidental Depressurization of the RCS	2841 MWt NSSS	No - See section III.1	References III.1, III.2, III.4	See section III.1	**Note 1**
(13) 15.2.13	Accidental Depressurization of the Main Steam System	N/A	N/A	No analysis, bounded by MSLB event	See section II.1	No analysis, bounded by MSLB event.
(14) 15.2.14	Inadvertent Operation of ECCS During Power Operation	2840.7 MWt NSSS	Yes	References II.8, II.10	See section II.1	Analysis performed using NRC approved methodologies.
(15) 15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate Emergency Core Cooling System	2831 MWt Core	Yes	References II.11, II.12, II.13, II.14, II.15	See section II.1	Analysis performed using NRC approved methodologies.
(16) 15.3.2	Minor Secondary System Pipe Breaks	N/A	N/A	No analysis, bounded by MSLB event	See section II.1	No analysis, bounded by MSLB event.

RIS 2002-03 Farley Units 1 and 2 FSAR Section	II.1.A Accident/Transient Title	II.1.B.i Power Level Used (MWt)	II.1.B.i Is Power Bounding for MUR? (Yes/No)	II.1.B.ii Approved by NRC or Conducted Using Methods / Processes Approved by the NRC	II.1.C Confirm That Bounding Event Determinations Remain Valid	II.1.D NRC Approval
(17) 15.3.3	Inadvertent Loading of a Fuel Assembly Into an Improper Position	Reload dependent	Yes	References II.17, II.18	See section II.1	Analysis performed using NRC approved methodologies.
(18) 15.3.4	Complete Loss of Forced Reactor Coolant Flow	2841 MWt NSSS	No - See section III.1	References III.1, III.2, III.3, III.4	See section III.1	**Note 1**
(19) 15.3.5	Waste Gas Decay Tank Rupture	2831 MWt Core	Yes	Reference II.32	See section II.1	NRC approval in Reference II.32
(20) 15.3.6	Single RCCA Withdrawal at Full Power	Reload dependent	Yes	References II.16, II.18, II.22, II.23	See section II.1	Analysis performed using NRC approved methodologies.
(21) 15.4.1	Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accidents)	2831 MWt Core	Yes	Reference II.24	See section II.1	Analysis performed using NRC approved methodologies.
(22) 15.4.2	Major Secondary System Pipe Rupture	2841 MWt NSSS (15.4.2.1) **(Note 2)** 2841 MWt NSSS (15.4.2.2)	Yes	References II.25, II.26, II.27 for 15.4.2.1 References III.4, III.6, III.7 for 15.4.2.2	See Section II.1 for 15.4.2.1 See section III.1 for 15.4.2.2	Analysis performed using NRC approved methodologies. **Note 1 for 15.4.2.2**
(23) 15.4.3	Steam Generator Tube Rupture	2775 MWt Core **(Note 4)**	Yes **(Note 4)**	Reference II.19	See section II.1	NRC approval in Reference II.19
(24) 15.4.4	Single Reactor Coolant Pump Locked Rotor	2841 MWt NSSS	No - See section III.1	References III.2, III.3, III.4	See section III.1	**Note 1**

RIS 2002-03 Farley Units 1 and 2 FSAR Section	**II.1.A** Accident/Transient Title	**II.1.B.i** Power Level Used (MWt)	**II.1.B.i** Is Power Bounding for MUR? (Yes/No)	**II.1.B.ii** Approved by NRC or Conducted Using Methods / Processes Approved by the NRC	**II.1.C** Confirm That Bounding Event Determinations Remain Valid	**II.1.D** NRC Approval
(25) 15.4.5	Fuel Handling Accident	2831 MWt Core	Yes	Reference II.19	See section II.1	NRC approval in Reference II.19
(26) 15.4.6	Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)	2831 MWt Core	Yes	References III.4, III.8, III.9, III.10	See section III.1	**Note 1**
(27) 15.5	Anticipated Transients without Scram (ATWS)	2841 MWt NSSS	No - See section III.1	References III.2, III.11, III.12, III.13	See section III.1	**Note 1**
(28) 6.2	Containment Analyses	2830.5 MWt Core (LOCA) 2841 MWt NSSS (SLB)	Yes	References II.28, II.29 (LOCA M&E Long Term) Reference II.30 (LOCA M&E Short Term) References II.8, II.31 (SLB M&E Long Term) Reference II.32 (Containment Response)	See section II.1	Analysis performed using NRC approved methodologies. NRC approval in Reference II.32.
(29) 3.11	Equipment Qualification	N/A	N/A	See section II.1	See section II.1	See section II.1
(30) 3.11	Flooding	N/A	N/A	See section II.1	See section II.1	See section II.1

RIS 2002-03 Farley Units 1 and 2 FSAR Section	II.1.A Accident/Transient Title	II.1.B.i Power Level Used (MWt)	II.1.B.i Is Power Bounding for MUR? (Yes/No)	II.1.B.ii Approved by NRC or Conducted Using Methods / Processes Approved by the NRC	II.1.C Confirm That Bounding Event Determinations Remain Valid	II.1.D NRC Approval
(31) 9.1.3	Spent Fuel Pool Loss of Cooling	2831 MWt Core	Yes	See section II.1	See section II.1	See section II.1
(32) 5.5.2.3.2	Natural Circulation Cooldown	2900 MWt Core	Yes	See section II.1	See section II.1	See section II.1
(33) APP 3J APP 3K	High Energy Line Breaks Outside Containment	2840.7 MWt NSSS	Yes	See section II.1	See section II.1	See section II.1
(34) 4	Fuel Evaluations Nuclear Design (ND) Fuel Rod Design (FRD) Core Thermal-Hydraulic Mechanical Design	2831 MWt Core (ND, FRD, Mechanical Design) 2823 MWt Core (DNBR)	Yes	Reference II.16 (ND) Reference III.4 (FRD) References III.1, III.3, III.16, III.25 (Core Thermal-Hydraulic)	See section II.1 for ND and Mechanical Design See section III.1 for FRD and Core Thermal-Hydraulic	Analysis performed using NRC approved methodologies for ND and Mechanical Design. **Note 1 for FRD and Core Thermal-Hydraulic.**

Notes:

1. Analysis discussed in Section III.1 will be incorporated as the AOR under 10 CFR 50.59 for implementation of the MUR-PU.
2. Analysis statepoints confirmed to remain valid and conservative with MUR-PU and PAD5 effects considered.
3. Uncontrolled boron dilution is not substantially impacted by the MUR-PU or PAD5 implementation, but a reanalysis was performed to simplify the AOR by consolidating existing calculations and evaluations.
4. An approximate 10 percent increase in the post-trip steam releases was added for SGTR, and this 10 percent increase more than offsets the 2 percent MUR-PU power increase.

III ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD DO NOT BOUND PLANT OPERATION AT THE PROPOSED UPRATED POWER LEVEL

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

RESPONSE:

See Section II.1 and Table II.1-1, Items 1 through 34, for discussion of the Farley Units 1 and 2 FSAR Chapter 15 accident analyses as well as other analyses that support licensing of the plant. All Farley analyses of record support the MUR-PU as described in Section II.1 except as discussed below.

The DNB events discussed in this section are primarily analyzed with statistical methods based on Reference III.1, which describes the RTDP for predicting the DNBR design limit in Westinghouse pressurized water reactors (PWRs) for use in analyzing DNB-related non-LOCA events. With the RTDP methodology, uncertainties in plant initial conditions and other factors are statistically combined to obtain the design limit DNBR that satisfies the 95/95 DNB design criterion. As such, the plant safety analyses are performed using the parameter values for the initial conditions (such as power, temperature, pressure, and RCS flow), without uncertainties. Because the core power level in the AOR for these events does not bound the increased core power level for the MUR-PU, these events were reanalyzed. The effects of PAD5 implementation have also been incorporated in the analyses in this section.

The non-LOCA analyses that included PAD5 for the Farley Units 1 and 2 MUR-PU satisfy the applicable limitations and conditions identified in the NRC Final Safety Evaluation for WCAP-17642-P/NP, Revision 1 (Reference III.4) as discussed below.

a) Farley Units 1 and 2 MUR-PU operating conditions comply with the specified cladding material, fuel properties, and reactor design features.

b) The PAD5 fuel temperature results used in the safety analyses for Farley Units 1 and 2 under MUR-PU operating conditions do not exceed the fuel melting temperature as calculated by PAD5.

c) There is no mention of the suggested response to RAI-22 in the revised TR. Therefore, it was not used for the Farley Units 1 and 2 non-LOCA analyses.

d) The model and methods improvement process (MMIP) was not approved by the NRC. Therefore, the MMIP was not used for the Farley Units 1 and 2 non-LOCA analyses.

e) This limitation is for required Westinghouse actions for issuance of the final accepted version (-A) of the Topical Report and subsequent actions to demonstrate and document, in

a letter addressed to the NRC, the continued applicability of PAD5 every 10 years starting in 2017. Therefore, this limitation is not applicable to the Farley Units 1 and 2 non-LOCA analyses.

Discussion of RIS 2002-03 Section III.1 Events

2. Uncontrolled RCCA Bank Withdrawal at Power – FSAR Section 15.2.2

   An uncontrolled RCCA withdrawal at power causes an increase in the core heat flux and may result from faulty operator action or a malfunction in the rod control system. Because the heat extraction from the SG lags behind the core power generation until SG pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in a violation of the DNBR design basis. Therefore, to avert damage to the fuel clad, the RPS is designed to terminate any such transient before the DNBR falls below the limit value or the fuel rod linear heat generation rate (HGR) limit is exceeded. The RPS and pressurizer safety valves (PSVs) are designed to preclude exceeding the RCS pressure boundary safety limit. The PSVs are required to provide overpressure protection. The PSVs, which have water filled loop seals, open to allow steam relief and RCS pressure relief when the pressurizer pressure exceeds the respective PSV set pressure. The main steam safety valves (MSSVs) open to allow secondary pressure relief, thus increasing the heat removal capability of the secondary side when the SG pressure exceeds the respective MSSV set pressure.

   For overpressure concerns, uncertainties about the initial conditions (i.e., power, temperature, pressure, and flow) are explicitly modeled. The AOR RCS overpressure cases modeled a range of initial NSSS power levels up to 2840.7 MWt, which is consistent with the maximum MUR-PU NSSS power. Therefore, the overpressure analyses are unaffected by the trade-off between the increased power level and decreased uncertainty. The overpressure case is also not sensitive to input changes to model TCD with burnup; thus, the overpressure analyses are not impacted by the MUR-PU.

   For DNB cases, the RTDP is used. The DNB cases were reanalyzed for the MUR-PU using the LOFTRAN code (Reference III.2). For the limiting cases, the VIPRE-W (VIPRE) computer code (Reference III.3) was used to calculate the minimum DNBR based on the transient statepoints from the output of the LOFTRAN transient runs. VIPRE-W is the Westinghouse version of the VIPRE-01 code, which was accepted by the NRC for application in Westinghouse analyses in Reference III.3. The DNB cases were analyzed with 2841 MWt NSSS (2831 MWt core) bounding power levels in LOFTRAN. For the limiting DNB cases, analysis statepoints were generated by LOFTRAN as input to the VIPRE code for the DNB calculations. A core power of 2823 MWt was applied in the VIPRE DNBR calculations for these limiting cases. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

   For the MUR-PU and PAD5 implementation uncontrolled RCCA bank withdrawal at power reanalysis, the method of analysis was unchanged from that applied in the AOR described in FSAR Section 15.2.2. The LOFTRAN computer code inputs were revised to account for the MUR-PU and PAD5 implementation for the DNB case. The high neutron flux and OTDT

reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates, initial power levels, and considering an end-of-cycle T_{avg} coastdown (i.e., the analysis demonstrated that the DNB design basis is met for all cases). In terms of minimum DNBR, the cases initialized at the high end of the T_{avg} window are more limiting than the T_{avg} coastdown cases.

The following items summarize the main acceptance criteria associated with this event.

1. The fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the DNBR design basis value for the entire transient.
2. Fuel integrity shall be maintained by ensuring that the maximum core power (heat flux) does not exceed the prescribed limit at any time during the transient.
3. Pressure in the RCS and MSS shall be maintained below 110 percent of the design values. Peak primary pressure results, documented for the AOR, remain valid for the MUR-PU.
4. An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently. This criterion is met by ensuring that the pressurizer does not reach a water-solid condition. Pressurizer filling (water solid) is not a concern for this event because the high pressurizer water level reactor trip will trip the reactor if the pressurizer approaches a filled condition. For post-reactor-trip considerations, the event is bounded by the loss of normal feedwater (LONF) event.
5. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding. This criterion is met by demonstrating that the DNB design basis is satisfied.

The evaluation of the RCS overpressure case concluded that the high pressurizer pressure, high neutron flux, and power range high positive neutron flux rate trip functions along with the PSVs and SG safety valves provide adequate protection over the entire range of possible reactivity insertion rates and initial conditions analyzed. Additionally, cases initiated at the high end of the T_{avg} window are conservative for the overpressure case. As such, the integrity of the RCS pressure boundary is maintained as the maximum transient pressure does not exceed the RCS pressure boundary safety limit.

Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the uncontrolled RCCA withdrawal at power transient analysis.

4. Uncontrolled Boron Dilution – FSAR Section 15.2.4

UBD is not substantially impacted by the MUR-PU or PAD5 implementation, but a reanalysis was performed to simplify the AOR by consolidating existing calculations and evaluations.

The UBD event is analyzed to demonstrate that there is sufficient time for plant operators to take necessary actions after a UBD begins and before all shutdown margin is lost, for Plant

Operating Modes 1 (at power), 2 (startup), and 6 (refueling). For the hot and cold shutdown operating modes (Modes 4 and 5), a plant operating procedure (Reference III.5) is applied by plant operators to identify the appropriate reactor coolant boron concentration that will conservatively assure sufficient time will be available to terminate a UBD event prior to the reactor reaching a critical condition. This plant operating procedure is based on a generic boron dilution analysis that bounds the Farley units and is applicable for dilution flow rates up to 300 gpm and at least 1000 gpm of residual heat removal (RHR) system flow. The UBD event analyses for Modes 1, 2, and 6, and the generic analysis basis of the plant operating procedure for Modes 4 and 5 were evaluated with respect to the MUR-PU and PAD5 implementation.

The NSSS power level of 2841 MWt was used for the analysis. The active mixing volumes were reviewed relative to the latest plant data for Farley, and the Mode 1, Mode 2, and Mode 6 cases active mixing volumes were captured in the updated UBD analysis. Regarding the generic analysis basis of the plant operating procedure for the shutdown modes, it was confirmed that the generic active mixing volumes were bounding relative to the latest plant data for Farley. The method of analysis for the UBD event is unchanged from that applied in the AOR described in FSAR Section 15.2.4. Specifically, hand calculations are performed for Modes 1, 2, and 6 to determine the amount of time available to plant operators before all shutdown margin is lost. The generic analysis basis of the plant operating procedure for the shutdown modes was determined to not be impacted by the MUR-PU, PAD5 implementation, and latest plant data.

Consistent with the AOR described in FSAR Section 15.2.4, the amount of time available to plant operators before all shutdown margin is lost must be greater than or equal to the following acceptance criteria:

- Mode 1 (Power Operation): 15 minutes
- Mode 2 (Startup): 15 minutes
- Mode 6 (Refueling): 18 minutes

With respect to the MUR-PU and PAD5 implementation, only the reactor trip time for the Mode 1 manual rod control case was potentially affected. To account for this impact and to update the other UBD cases for convenience, new operator action times were calculated for Modes 1 (with automatic and manual rod control), 2, and 6. The analysis results demonstrate that the applicable acceptance criteria are satisfied. Based on these results the MUR-PU and PAD5 implementation are acceptable with respect to the UBD event analysis.

5. Partial Loss of Forced Reactor Coolant Flow – FSAR Section 15.2.5

A partial loss of reactor coolant flow event can result from a mechanical or electrical failure in an RCP, or from a fault in the power supply to the pump or pumps supplied by an RCP bus. If the reactor is at power at the time of the event, the immediate effect of the reactor coolant flow reduction is a rapid increase of the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

Initial reactor power, pressurizer pressure, and RCS average temperature (consistent with MUR-PU conditions) were modeled at the nominal full power value of 2823 MWt. The most

limiting single failure for a partial loss of flow event is the failure of a protection train. No single active failure will prevent the RPS from functioning properly.

The partial loss of flow transient was analyzed with two computer codes. First, the LOFTRAN computer code (Reference III.2) was used to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code (Reference III.3) was then used to calculate the heat flux and DNBR using the RTDP (Reference III.1) based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from the output of the LOFTRAN transient run. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup are conservatively accounted for in the analysis. The acceptance criterion of interest for the partial loss of flow analysis is that the DNB design basis must be satisfied.

The partial loss of flow event was reanalyzed for the MUR-PU and PAD5 implementation. The analysis adequately accounts for input changes associated with the MUR-PU and PAD5 implementation, and was performed using acceptable analytical models. The analysis results demonstrated that all applicable acceptance criteria are satisfied, and that the MUR-PU and PAD5 implementation are acceptable with respect to the partial loss of flow transient analysis.

7. Loss of External Electrical Load and/or Turbine Trip – FSAR Section 15.2.7

A TT event is bounding for loss of external load (i.e., LOL), loss of condenser vacuum, inadvertent closure of the main steam isolation valves (MSIVs), and other TT events. For a TT event, the turbine stop valves close rapidly (typically 0.1 second) on loss of trip fluid pressure actuated by one of a number of possible TT signals. Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the TT and initiate steam dump. The loss of steam flow results in an almost immediate rise in secondary system temperature. For a TT, the reactor would be tripped directly (unless below approximately 50 percent [P-9] power) on a signal from the turbine stop valves.

The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the AFWS to guarantee adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the SG safety valves may lift to provide pressure control.

Cases for both Farley Units 1 and 2 with Westinghouse Model 54F SGs, were analyzed at conditions designed to provide the most conservative and limiting results. The analysis used the maximum uprated NSSS power of 2841 MWt, including a nominal RCP heat addition of 10 MWt.

Multiple cases were analyzed to address specific acceptance criteria; specifically, minimum DNBR and maximum RCS and MSS pressure. Implementation of the PAD5 fuel performance analysis and design model (Reference III.4) required a full reanalysis of the maximum RCS pressure case and the minimum DNBR case. Additionally, an explicit MSS

overpressure analysis designed to maximize SG pressure was analyzed as part of the MUR-PU with PAD5 implementation. The analyses were performed with the NRC-approved RETRAN computer code (Reference III.6). Consistent with the PAD5 fuel performance code, the effects of fuel TCD with burnup are conservatively accounted for in the analysis.

To bound all the TT transients, the behavior of the unit was evaluated for a complete loss of steam load from full power primarily to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a reactor trip (i.e., no reactor trip on TT). No credit was taken for steam dump. Main feedwater flow is terminated at the time of TT, with no credit taken for AFW to mitigate the consequences of the transient.

The acceptance criteria of interest for the LOL/TT transient are as follows:

1. RCS and MSS pressures shall be maintained below 110 percent of design values.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs.
3. An incident of moderate frequency shall not generate a more serious plant condition without other faults occurring independently. This criterion is met by ensuring that the pressurizer does not reach a water-solid condition.
4. An incident of moderate frequency, in combination with any single active component failure or single operator error, shall be considered an event for which an estimate of the number of potential fuel failures will be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below the values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding. This criterion is met by demonstrating that the DNB design basis is satisfied, and the RCS pressure limit is satisfied.

In accordance with current licensing basis as approved by the NRC in Reference III.26, the dose consequences of the bounding event that addresses all three events (Loss of Offsite Power (LOSP), LOL, and TT) must remain within 10 percent of the 10 CFR 100 limits.

The MUR-PU evaluation demonstrates that the iodine concentrations in the environmental releases are controlled by the TSs and are therefore unaffected by the MUR-PU, and the noble gas concentrations (based on 1-percent fuel defects assuming a core thermal power level of 2831 MWt) remain applicable to the MUR-PU operations. In addition, the analysis of record includes sufficient margin in the secondary steam mass environmental releases to bound the increased values applicable for MUR-PU operation. Therefore, the results and conclusions presented in FSAR Section 15.2.7 remain valid at MUR-PU conditions. Therefore, the MUR-PU will have no significant effect on the LOL/TT/LOSP dose consequences reported in FSAR Chapter 15.

The LOL/TT transient was reanalyzed for the MUR-PU and PAD5 implementation. The analysis adequately accounts for input changes associated with the MUR-PU and PAD5 implementation and was performed using approved analytical models. The analysis results demonstrate that all applicable acceptance criteria are satisfied. Based on these results, the

MUR-PU and PAD5 implementation are acceptable with respect to the LOL/TT transient analysis.

8. Loss of Normal Feedwater – FSAR Section 15.2.8

The LONF event was analyzed for the MUR-PU and PAD5 implementation to incorporate the associated fuel temperature data. A LONF (from pump failures, valve malfunctions, or loss of offsite AC power) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If the reactor is not tripped during this event, core damage would possibly occur as a result of the loss of heat sink while at power. If an alternative supply of feedwater is not supplied to the plant, residual heat following a reactor trip may heat the primary system water to the point where water relief from the pressurizer could occur. A significant loss of water from the RCS could lead to core uncovery and subsequent core damage. However, because a reactor trip occurs well before the SG heat transfer capability is reduced, the primary system conditions never approach those that would result in a DNB condition.

The analysis used an NSSS power of 2841 MWt with all three RCPs in operation providing a constant reactor coolant volumetric flow equal to the thermal design flow (TDF) value. A conservatively high RCP heat addition of 15 MWt (5 MWt/pump) was assumed. It was assumed that the operator manually trips two of three RCPs 10 minutes after reactor trip (rod motion). At this time, the RCP heat addition is reduced from 15 to 5 MWt.

A detailed analysis using the Westinghouse RETRAN-02 computer code model (References III.6 and III.7) was performed to determine the plant transient conditions following an LONF. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer power-operated relief valve (PORVs), heaters, sprays, SGs, MSSVs, and the AFWS; and computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG level and mass, and reactor coolant average temperature. Following the method approved in Reference III.7 the analysis takes credit for a more realistic initial SG water mass than predicted with the models approved in Reference III.6. Consistent with Reference III.7, the SG water masses were reduced to provide conservatism. As approved in Reference III.7, the analysis used the SG mass as the trip parameter instead of the less accurate SG level determined by the RETRAN-02 model. The analysis did not credit the enhanced thick metal heat transfer model approved in Reference III.7 because it was determined that acceptable results, with sufficient margin to the limit, were obtained. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

The following items summarize the acceptance criteria associated with this event:

1. The critical heat flux (CHF) shall not be exceeded. This is demonstrated by precluding DNB.
2. Pressure in the RCS and the MSS shall be maintained below 110 percent of the design pressures.
3. There shall be no propagation to a more serious event.

With respect to DNB and RCS / MSS overpressurization, the LONF event is bounded by the loss of external electrical load event. For ease in interpreting the transient results following a

LONF, the following restrictive acceptance criterion is used: the pressurizer shall not become water solid.

With respect to DNB, the LONF event is bounded by the LOL event, which demonstrates that the minimum DNBR is greater than the safety analysis DNB acceptance criterion. With respect to RCS and MSS overpressurization, the LONF event is bounded by the loss of external electrical load event, which demonstrates that the peak primary and secondary-side pressures remain below 110 percent of design at all times.

The analysis showed that following a LONF, the AFWS is capable of removing the stored and residual heat thus preventing overpressurization of the RCS, overpressurization of the secondary side, water relief through the PSVs, and uncovering of the reactor core. The analysis showed that the pressurizer does not reach a water solid condition. Therefore, the LONF event does not adversely affect the core, the RCS, or the MSS.

Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the LONF analysis.

9. Loss of Offsite Power (LOSP) to the Station Auxiliaries – FSAR Section 15.2.9

The loss of all AC power to the station auxiliaries event was analyzed for the MUR-PU and PAD5 implementation to incorporate the associated fuel temperature data. The first few seconds after a loss of AC power to the RCPs closely resembles the analysis of the complete loss of forced reactor coolant flow event in that the RCS would experience a rapid flow reduction transient. This aspect of the loss of AC power event is bounded by the analysis performed for the complete loss of flow event which demonstrates that the DNB design basis is met. The analysis of the loss of AC power event demonstrates that RCS natural circulation and the AFWS are capable of removing the stored and residual heat, and consequently will prevent RCS or MSS overpressurization and core uncovery.

The analysis was performed at an NSSS power of 2841 MWt with all three RCPs in operation providing a constant reactor coolant volumetric flow equal to the TDF value. A nominal RCP heat addition of 10 MWt was assumed. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup are conservatively accounted for in the analysis.

A detailed analysis using the Westinghouse RETRAN-02 computer code model (References III.6 and III.7) was performed to determine the plant transient conditions following a loss of all AC power to the station auxiliaries. The code models the core neutron kinetics, RCS including natural circulation, pressurizer, pressurizer PORVs, heaters, sprays, SGs, MSSVs, and the AFWS. The code computes pertinent variables, including the pressurizer pressure, pressurizer water level, SG level and mass, and reactor coolant average temperature. Following the method approved in Reference 2, the analysis takes credit for a more realistic initial SG water mass than predicted with the models approved in Reference III.6. Consistent with Reference III.7, the SG water masses were reduced to provide conservatism. As approved in Reference III.6, the analysis used the SG mass as the trip parameter instead of the less accurate SG level determined by the RETRAN-02 model. The analysis did not credit the enhanced thick metal heat transfer model approved in Reference III.7 because it was determined that acceptable results, with sufficient margin to the limit, were obtained.

The following items summarize the acceptance criteria associated with this event:

1. The CHF shall not be exceeded. This is demonstrated by precluding DNB.
2. Pressure in the RCS and MSS shall be maintained below 110 percent of the design pressures.
3. There shall be no propagation to a more serious event.

With respect to DNB, the loss of all AC power to the station auxiliaries event is bounded by the complete loss of forced reactor coolant flow event. With respect to RCS and MSS overpressurization, the loss of all AC power to the station auxiliaries event is bounded by the loss of external electrical load event. For ease in interpreting the transient results following a loss of all AC power to the station auxiliaries event, the following restrictive acceptance criterion has been used: the pressurizer shall not become water solid.

In accordance with current licensing basis as approved by the NRC in Reference III.26, the dose consequences of the bounding event that addresses all three events (LOSP, LOL, and TT) must remain within 10 percent of the 10 CFR 100 limits.

Refer to Section III.1, Item 7, "Loss of External Electrical Load and /or Turbine Trip," which summarizes the basis of the conclusion that the MUR-PU will have no significant effect on the LOL/TT/LOSP dose consequences reported in FSAR Chapter 15. With respect to DNB, the loss of all AC power to the station auxiliaries event is bounded by the complete loss of flow event, which demonstrates that the minimum DNBR is greater than the safety analysis DNB acceptance criterion. With respect to RCS and MSS overpressurization, the loss of all AC power to the station auxiliaries event is bounded by the loss of external electrical load event, which demonstrates that the peak primary and secondary system pressures remain below 110 percent of design at all times. The results of the analysis show that the pressurizer does not reach a water solid condition. Therefore, the LOSP event does not adversely affect the core, the RCS, or the MSS.

Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the loss of offsite power to the station auxiliaries analysis.

10. Excessive Heat Removal Due to Feedwater System Malfunctions – FSAR Section 15.2.10

A change in SG feedwater conditions that results in an increase in feedwater flow and/or a decrease in feedwater temperature could result in excessive heat removal from the plant primary coolant system. An accidental opening of a feedwater bypass valve, which diverts flow around a portion of the feedwater heaters (FWHs), is an event that causes a reduction in feedwater inlet temperature to the SGs. An accidental full opening of one feedwater control valve would cause excessive feedwater flow to one or more of the SGs. Both reduced feedwater temperature and increased feedwater flow are feedwater system malfunctions that produce increased subcooling in the affected SGs.

At power, this increased subcooling will create a greater load demand on the RCS with a resulting decrease in RCS temperature. In the presence of a negative moderator temperature coefficient (MTC), the decrease in RCS temperature will produce a positive reactivity insertion. The thermal capacity of the secondary plant and of the RCS attenuates the increase in core power from these reductions in feedwater temperature. The

overpower/overtemperature protection systems (neutron overpower, overtemperature and overpower ΔT (OTΔT / OPΔT) trips) are designed to prevent any power increase that could lead to a DNBR less than the limit value.

In addition to the overpower/overtemperature protection systems, the SG high-high level trip, which closes the feedwater valves, is a protection function credited for mitigating the consequences of a feedwater system excessive flow malfunction.

Two cases were analyzed:

1. Accidental opening of one feedwater control valve with the reactor in manual rod control at full power, and,
2. Accidental opening of one feedwater control valve with the reactor in automatic rod control at full power.

The accident was analyzed at an initial NSSS power of 2841 MWt. Although not sensitive to the impact, the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

The feedwater system malfunction analysis uses the NRC-approved LOFTRAN computer code (Reference III.2) and RTDP methodology (Reference III.1) to calculate the minimum DNBR. The HFP cases were analyzed, which model the opening of one feedwater control valve that increases main feedwater flow from the nominal full power value to 184 percent of the nominal full power value. Cases with automatic and manual rod control were analyzed. This failure produces the greatest thermal load on the primary system and subsequent positive reactivity insertion rate.

For the feedwater control valve accident at zero-load condition, a feedwater valve malfunction could occur that results in an increase in flow to one SG from zero to the nominal full-load value for one SG. The addition of cold feedwater may also cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative MTC of reactivity. However, the cooldown and resulting positive reactivity insertion associated with this event would be much less than for the rupture of a main steam line event analyzed at zero load conditions (HZP SLB). The no-load transient is also less severe than the full-power case.

The analysis for the feedwater system malfunction is performed to confirm that the DNB design basis is satisfied. In addition, the analysis must confirm that the peak linear heat rate (typically expressed in kW/ft) does not exceed the limit value that precludes fuel centerline melting. This is demonstrated by showing that the peak core average power remains below 120 percent of the nominal value.

The most limiting case for a feedwater system malfunction is at HFP with manual rod control. This case was only slightly more limiting than the case with automatic rod control.

The feedwater malfunction transient was reanalyzed (limiting HFP cases) and evaluated (no-load case and feedwater temperature reduction cases) for the MUR-PU and PAD5 implementation. The analysis and evaluation adequately account for input changes associated with the MUR-PU and PAD5 implementation, and was performed using acceptable analytical models. The HFP analysis demonstrates that the DNB design basis is

met for both cases. In terms of minimum DNBR, the full power manual rod control case was slightly more limiting than the automatic rod control case. Both cases demonstrated that core average power remained well below 120 percent, thus ensuring that the peak linear heat rate criterion is also met. The feedwater control valve malfunction at no-load conditions is less limiting than the full power case. The results of a no-load case would be bounded by the HZP SLB analysis. The feedwater temperature reduction case is bounded by the excessive load increase analysis.

Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the excessive heat removal due to feedwater system malfunctions transients.

12. Accidental Depressurization of the RCS – FSAR Section 15.2.12

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer PORV or safety valve. Since a PSV is designed to relieve approximately twice the steam flow rate of a PORV, thereby allowing for a much more rapid depressurization upon opening, the most-severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a PSV.

Initially, the event results in a rapidly decreasing RCS pressure which could reach hot leg saturation conditions without RPS intervention. If saturated conditions are reached, the rate of depressurization is slowed considerably; however, the pressure continues to decrease throughout the event. The effect of the pressure decrease is to increase power via the moderator density feedback; however, if the plant is in the automatic mode, the rod control system functions to maintain the power essentially constant throughout the initial stages of the transient. The average coolant temperature remains approximately the same, but the pressurizer level increases until reactor trip because of the decreased reactor coolant density.

The accidental depressurization of the RCS analysis for Farley Units 1 and 2 used the NRC-approved LOFTRAN computer code (Reference III.2) and RTDP methodology (Reference III.1) to calculate a minimum DNBR. The accident was analyzed at 2841 MWt NSSS power. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

The criterion of interest for the accidental depressurization of the RCS analysis, which conservatively models the inadvertent opening of a PSV, is that the DNB design basis is satisfied. The results of the analysis show that the pressurizer low pressure and OTΔT RPS signals provide adequate protection against the RCS depressurization event. Thus, there will be no cladding damage or release of fission products to the RCS.

Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the excessive heat removal due to accidental depressurization of the RCS transients.

18. Complete Loss of Forced Reactor Coolant Flow – FSAR Section 15.3.4

A complete loss of forced reactor coolant flow event may result from a simultaneous loss of electrical supplies to all RCPs. If the reactor is at power at the time of the event, the immediate effect of the reactor coolant flow reduction is a rapid increase of the coolant

temperature. This increase could result in a DNB with subsequent fuel damage if the reactor is not tripped promptly.

The following complete loss of forced reactor coolant flow cases were analyzed using the RTDP (Reference III.1):

1. Coastdown of all three RCPs with three loops initially in operation
2. Frequency decay event resulting in a complete loss of forced reactor coolant flow

An NSSS power of 2841 MWt was used in the analysis. The most limiting single failure for a complete loss of flow event is the failure of a protection train. No single active failure will prevent the RPS from functioning properly.

The transients were analyzed with two computer codes. First, the LOFTRAN computer code (Reference III.2) was used to calculate the loop and core flows during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The VIPRE computer code (Reference III.3) was then used to calculate the heat flux and DNBR using the RTDP based on the nuclear power and RCS temperature (enthalpy), pressure, and flow from the output of the LOFTRAN transient run. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

The acceptance criterion of interest for the complete loss of flow analysis is that the DNB design basis must be satisfied. The complete loss of flow event was reanalyzed for the MUR-PU and PAD5 implementation. The analysis adequately accounts for input changes associated with the MUR-PU and PAD5 implementation, and was performed using acceptable analytical models. The analysis results demonstrate that all applicable acceptance criteria are satisfied, and that the MUR-PU and PAD5 implementation are acceptable with respect to the complete loss of flow transient analysis.

22. Major Secondary System Pipe Rupture – FSAR Section 15.4.2.2

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell-side fluid inventory in the SGs. A feedline rupture reduces the ability to remove heat generated by the core from the RCS. The AFWS is designed to guarantee that adequate feedwater will be available to provide decay heat removal.

An NSSS power of 2841 MWt and maximum RCP heat (15 MWt) was applied in the case that models offsite power always available, while the nominal RCP heat (10 MWt) was applied in the feedwater line rupture cases that model a LOSP. The transient response following a feedwater pipe rupture event was calculated by a detailed digital simulation of the plant. The analysis models a simultaneous loss of main feedwater to all SGs and subsequent reverse blowdown of the faulted SG, except for the limiting break size of 0.200 ft^2, which assumes that 30 percent of the initial main feedwater flow continues to the two intact SGs until the time of reactor trip (rod motion). A detailed analysis using the Westinghouse RETRAN-02 computer code model (References III.6 and III.7) was performed to determine the plant transient conditions following a feedwater system pipe rupture. The code models the core neutron kinetics, RCS, including natural circulation, pressurizer, SGs, SI system, and the AFWS. The code computes pertinent parameters, including the core

heat flux, RCS temperature and RCS pressure. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis.

The Standard Review Plan (Revision 1) requires that the specific criteria used in evaluating the consequences of the feedline rupture shall be:

1. Pressure in the RCS and MSS should be maintained below 110 percent of the design pressures.
2. Any fuel damage that may occur during the transient should be of a sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.
3. Any activity release must be such that the calculated doses at the site boundary are within a small fraction of the guidelines of 10 CFR Part 100.

Westinghouse applies a single, conservative acceptance criterion that no bulk boiling occurs in the primary coolant system following a feedline rupture prior to the time that the heat removal capability of the SGs being fed AFW exceeds NSSS residual heat generation. With bulk boiling precluded in the RCS, the core remains covered, thus precluding significant fuel damage due to dryout. With respect to DNB, this event is bounded by the MSLB and/or LOL / TT events. Since DNB is precluded in those events, fuel damage caused by DNB is also precluded for feedline rupture. As such, the core remains in a coolable geometry throughout the event.

Results of the analysis showed that for the postulated feedline rupture, the applied AFWS capacity is adequate to remove core decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the fuel assemblies in the reactor core. Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the major feedwater line rupture event.

24. Single Reactor Coolant Pump Locked Rotor – FSAR Section 15.4.4

A transient analysis was performed for the instantaneous seizure of a RCP rotor (locked rotor (LR)). Flow through the affected RCL is rapidly reduced, leading to a reactor trip on a low flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. At the same time, heat transfer to the shell side of the SG is reduced, first because the reduced flow results in a decreased tube side film coefficient, and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of coolant in the reactor core, combined with the reduced heat transfer in the SG, causes an insurge into the pressurizer and a pressure increase throughout the RCS. The pressure increase actuates the automatic pressurizer spray system, opens the PORVs, and opens the PSVs. The sequence of events initiated by the insurge depends on the rate of insurge and pressure increase. The PORVs are designed for reliable operation and would be expected to function properly during the event. However, for conservatism, their pressure-reducing effect and the pressure reducing effect of the pressurizer spray were not included in this analysis.

The consequences of a LR (i.e., an instantaneous seizure of a pump shaft) are very similar to those of a pump shaft break. The initial rate of the reduction in coolant flow is slightly greater for the LR event. However, with a broken shaft, the impeller could conceivably be free to spin in the reverse direction. The effect of reverse spinning is to decrease the steady-state core flow when compared to the LR scenario. Only one analysis, which permits reverse spinning but no forward flow, has been performed and represents the most limiting condition for the LR and pump shaft break events.

Two cases were examined for the LR event. The first case focused on maximizing the primary system pressure, fuel clad temperature, and zirconium-water reaction. This case is referred to as the peak pressure / PCT case. The second case determined the percentage of fuel rods that experience a DNBR less than the limit value. This case is referred to as the rods-in-DNB case. Two digital computer codes were used to analyze this event. The LOFTRAN code (Reference III.2) was used to calculate the resulting loop and core coolant flow following the pump seizure, the time of reactor trip, nuclear power during the event and peak RCS pressure. The VIPRE code (Reference III.3) was used to calculate the PCT and rods-in-DNB using the nuclear power, RCS temperature, primary pressure and reactor coolant flow from LOFTRAN. Consistent with the PAD5 fuel performance code (Reference III.4), the effects of fuel TCD with burnup were conservatively accounted for in the analysis. An NSSS power of 2841 MWt was used in the analysis. The most limiting single failure for a LR event is the failure of a protection train. No single active failure will prevent the RPS from functioning properly.

No credit was taken for the pressure-reducing effect of the pressurizer PORVs, pressurizer spray, or steam dump system. Although these systems are expected to function and would result in a lower peak RCS pressure, an additional degree of conservatism was provided by not including their effect.

The film boiling coefficient was calculated in the VIPRE code using the Bishop-Sandberg-Tong film boiling correlation (Reference III.3). The fluid properties were evaluated at the film temperature. The program calculated the film coefficient at every time step based upon the actual heat transfer conditions at the time. The nuclear power, system pressure, bulk density, and RCS flow rate as a function of time were based on the LOFTRAN results.

The zirconium-steam reaction can become significant above 1,800°F (cladding temperature). The Baker-Just parabolic rate equation (Reference III.3). was used to define the rate of zirconium-steam reaction. The effect of the zirconium-steam reaction was included in the calculation of the fuel cladding temperature transient.

The acceptance criteria associated with this event are summarized as follows:

1. Fuel cladding damage, including melting, due to the increased reactor coolant temperatures must be prevented. This is demonstrated by showing that the maximum cladding temperature at the core hot spot remains below 2,700°F (for ZIRLO® cladding material) and 2,375°F (for Optimized ZIRLO™ High Performance Fuel Cladding

Material)[1], and the zirconium-water reaction at the core hot spot is less than 16 percent by weight.

2. Pressures in the RCS are to be maintained less than 110 percent of the RCS design pressure.
3. The total number of rods-in-DNB calculated for the associated dose analysis is less than 20 percent of the core.

In accordance with the current licensing basis which incorporates the AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following the LR Accident DBA meet the onsite and offsite dose limits in 10 CFR 50.67 and RG 1.183, Revision 0. It was determined that the limit of 20 percent rods in DNB for the LR Accident remains valid for the MUR-PU. The inventory of radionuclides in the fuel gap is currently based on a core thermal power of 2831 MWt (102 percent of current RTP of 2775 MWt), which bounds operation at MUR-PU operating conditions. Additional factors that can affect the fuel gap inventory are fuel enrichment and burnup, both of which remain unchanged by the MUR-PU. In addition, the evaluation confirmed that the $F_{\Delta H}$ limit (maximum peaking factor) of 1.7 will not be challenged, and that the maximum linear heat generation rate will not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU at MUR-PU conditions. The above confirmed the continued applicability of the non-LOCA gap fractions presented in Table 3 of RG 1.183 for MUR-PU conditions. In addition, the analysis of record includes sufficient margin in the secondary steam mass environmental releases to bound the increased values applicable for MUR-PU operation. Therefore, the MUR-PU will have no significant effect on the dose consequences at the EAB, LPZ, and in the CR for DBAs reported in FSAR Chapter 15.

The LR/shaft break transient was reanalyzed for the MUR-PU and PAD5 implementation. The analysis adequately accounts for input changes associated with the MUR-PU and PAD5 implementation and was performed using acceptable analytical models. The analysis demonstrates that all applicable acceptance criteria are satisfied. Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the LR/shaft break transient analysis.

26. Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection) – FSAR Section 15.4.6

An RCCA rod ejection event is defined as a mechanical failure of a CRDM pressure housing resulting in the ejection of the RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals. A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible as justified in FSAR Section 15.4.6 and Reference III.10.

[1] ZIRLO® and Optimized ZIRLO™ are trademarks or registered trademarks of Westinghouse Electric Company LLC, its Affiliates and/or its Subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

Although the rod ejection safety analysis described in FSAR Section 15.4.6 could have accommodated the MUR-PU, implementation of the PAD5 fuel performance analysis and design model (Reference III.4) requires a full reanalysis of the rod ejection event. PAD5 implementation affects the rod ejection analysis inputs of fuel average and fuel surface temperatures, fuel thermal conductivity, and fuel melt temperature, all of which vary as a function of fuel burnup. Whereas the fuel temperatures and thermal conductivity generally become more limiting with PAD5, the fuel melt temperature is less limiting. To help offset the impact of PAD5 implementation, margin available in the bounding full power ejected rod worth and post-ejection hot channel factor (FQ) values was utilized.

The initial core power was either 0 percent (for HZP cases) or 100 percent (for HFP cases) of the bounding MUR-PU core power of 2831 MWt. This uprated core power (with uncertainty) was a slight increase over the initial core power modeled in the HFP cases of the previous rod ejection analysis, which was 102 percent of 2775 MWt (2830.5 MWt). The TWINKLE computer code (Reference III.8) was used to perform average core channel calculations. The FACTRAN computer code (Reference III.9), was used to perform transient hot spot calculations. PAD5 (Reference III.4) incorporates TCD with burnup. Equation 6-4 in Section 6.1.2 of Reference III.4 provides the PAD5 thermal conductivity equation for UO_2 fuel with 95 percent theoretical density. Consistent with this equation, a UO_2 thermal conductivity versus fuel temperature curve was calculated for each UO_2 burnup considered and used as input to the FACTRAN calculations. A bounding burnup of 62,000 MWD/MTU was considered for the HFP cases, and a bounding burnup of 65,000 MWD/MTU was considered for the HZP cases.

Maximum initial fuel average and surface temperatures were used as input to FACTRAN for the full power rod ejection cases. Therefore, the new PAD5 maximum fuel temperatures for the 17×17 Vantage 5 fuel (with 0.360-inch outer diameter cladding) in use at Farley Units 1 and 2 were used to determine bounding fuel temperature values for the full power cases. Based on the steady-state hot channel factor of 2.50, the peak initial local rod power at full power corresponds to 13.89 kW/ft. For this rod power, the overall maximum fuel average temperature occurs at the maximum UO_2 burnup of 62,000 MWD/MTU considered in the PAD5 calculations. The revised fuel temperatures will be confirmed to remain bounding as part of the reload safety evaluation process for future MUR-PU cycles.

The fuel melting temperature was used as input to FACTRAN for determining the percentage of fuel melting in the hot spot calculations. With PAD5, an improved melting temperature model is used in which burnup has less of an impact on the fuel melt temperature limit, as indicated by the following equation that is conservatively based on the UO_2 fuel melting point equation used in PAD5 (Equation 6-14 in Section 6.1.5 of Reference III.4).

$$T_{melt} = 5080°F - \frac{9}{10000}(BU)$$

Where,

T_{melt} = UO_2 melting temperature, °F

BU = UO_2 burnup, MWD/MTU

Consistent with this new equation, new UO_2 melting temperatures were calculated for each UO_2 burnup considered and used as input to the FACTRAN calculations; a bounding burnup of 62,000 MWD/MTU was considered for the HFP cases and a bounding burnup of 65,000 MWD/MTU was considered for the HZP cases.

For the MUR-PU and PAD5 implementation rod ejection analysis, the method of analysis is unchanged from that applied in the AOR described in FSAR Section 15.4.6. TWINKLE and FACTRAN computer code inputs were revised to account for the MUR-PU and PAD5 implementation. As per FSAR Section 15.4.6, the calculations of the rod ejection transient analysis are performed in two stages, first an average core channel calculation and then a hot region calculation. The average core calculation, performed by the TWINKLE computer code, uses spatial neutron-kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Fuel enthalpy and temperature transients at the hot spot are then determined using the FACTRAN computer code, which multiplies the average core energy generation by the hot channel factor and performs a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to persist throughout the transient.

The applicable acceptance criteria for the rod ejection analysis are unchanged from those applied in the AOR described in FSAR Section 15.4.6, and are summarized as follows.

1. Average fuel pellet enthalpy at the hot spot must be maintained below 225 cal/g for unirradiated and 200 cal/g for irradiated fuel. The 200 cal/g (360 Btu/lbm) limit is applied to cover both unirradiated and irradiated fuel.
2. Peak RCS pressure must be less than that which could cause stresses in RCS components to exceed the faulted-condition stress limits.
3. The peak RCS pressure aspects of the rod ejection transient are addressed generically in Reference III.10, which concluded that the rod ejection transient is not limiting with respect to RCS pressure. The rod ejection analysis input changes associated with the MUR-PU and PAD5 implementation do not invalidate the generic RCS pressure conclusion, and therefore no additional analysis was performed to address the RCS pressure criterion.
4. Fuel melting must be limited to less than 10 percent of the fuel pellet volume at the hot spot, even if the average fuel pellet enthalpy is below the 200 cal/g (360 Btu/lbm) fuel enthalpy limit.

In accordance with the current licensing basis which incorporates the AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following the Control Rod Ejection Accident meet the onsite and offsite dose limits set by 10 CFR 50.67 and RG 1.183, Revision 0. The inventory of radionuclides released due to fuel cladding failure / melted fuel is currently based on a core thermal power of 2831 MWt (102 percent of current RTP of 2775 MWt), which bounds operation at MUR-PU operating conditions. Additional factors that can affect the fuel gap inventory are fuel enrichment and burnup, both of which remain unchanged by the MUR-PU. The evaluation confirmed that the $F_{\Delta H}$ limit (maximum peaking factor) of 1.7 will not be challenged. The CREA gap fractions are based on Note 11 of RG 1.183, Rev. 0, which remains applicable for MUR-PU

conditions. In addition, the analysis of record includes sufficient margin in the secondary steam mass environmental releases to bound the increased values applicable for MUR-PU operation. Therefore, the MUR-PU will have no significant effect on the dose consequences at the EAB and LPZ and in the CR for DBAs reported in FSAR Chapter 15.

The rod ejection transient was reanalyzed for the MUR-PU and PAD5 implementation. The analysis adequately accounts for input changes associated with the MUR-PU and PAD5 implementation, and was performed using acceptable analytical models. The analysis results demonstrate that all applicable acceptance criteria are satisfied. As such, there is no danger of either sudden fuel dispersal into the coolant or further consequential damage to the RCS, and LTCC will not be impaired. Based on these results, the MUR-PU and PAD5 implementation are acceptable with respect to the rod ejection transient analysis.

27. Anticipated Transients Without Scram (ATWS) – FSAR Section 15.5

An ATWS is an anticipated operational occurrence (such as a loss of feedwater, loss of condenser vacuum, or LOSP) that is accompanied by a failure of the reactor trip system to shut down the reactor. In the worst case, an unmitigated ATWS might result in RCS pressure that compromises the integrity of the RCS. The final ATWS rule (Reference III.11), requires Westinghouse-designed pressurized water reactors to incorporate a system diverse from the reactor trip system that automatically actuates the AFWS and initiates a TT for conditions indicative of an ATWS. The installation of the ATWS mitigating system actuation circuitry (AMSAC), described in FSAR Section 7.8, satisfies the final ATWS rule.

It must also be demonstrated that the deterministic ATWS analyses that form the basis for this rule and the AMSAC design remain valid for the plant. This is typically done by confirming that the analyses documented in Reference III.12 remain valid or by performing new deterministic analyses for the proposed plant state. To address the MUR-PU, the LOL and LONF ATWS events were reanalyzed to ensure that the analytical basis for the final ATWS rule continues to be met. The LOL and LONF ATWS events are the two most limiting RCS overpressure transients reported in Reference III.12. Consistent with Reference III.12, the analytical basis of the final ATWS rule is shown to be met for each ATWS case reanalyzed by demonstrating that the calculated peak RCS pressure remains less than the ASME (Reference III.13) Service Level C pressure limit of 3200 psig. Consistent with the reference analyses for the final ATWS rule analytical basis (Reference III.12), the NRC required that all parameters be BE values except for the MTC, which is to be a full power value that is bounding for at least 95 percent of a given cycle. The conditions of most significance to the resulting peak RCS pressure following the LOL and LONF ATWS events are the initial power level, total reactivity feedback (primarily MTC), primary-side pressure relief capacity, and the AMSAC actuation setpoint and delays.

Using the LOFTRAN computer code (Reference III.2), the two most limiting ATWS cases with respect to the peak RCS pressure as documented in Reference III.12, the LOL and LONF ATWS cases, were reanalyzed for the MUR-PU Program. An NSSS power level of 2841 MWt was used for the analysis. Other than the uprated NSSS power, the most significant changes that were made to the ATWS analysis inputs were to the AMSAC actuation timing. Rather than model the AMSAC-initiated TT at 30 seconds (LONF case only) and the AMSAC-initiated AFW at 90 seconds from event initiation as was done in the previous ATWS analyses, e.g., Reference III.12, the Farley-specific AMSAC low SG water

level setpoint was simulated along with the Farley-specific AMSAC delays for TT (2.5 seconds), AFW actuation (60 seconds), and AMSAC timer delay (10 seconds).

To remain compliant with the basis of the final ATWS rule (10 CFR 50.62), the ASME Service Level C pressure limit of 3200 psig (3214.7 psia) for the RCS must be met for at least 95 percent of the cycle. With this peak RCS pressure limit met and the applied MTC confirmed for at least 95 percent of a cycle, the ATWS analysis results are acceptable. The ATWS MTC is met for the anticipated operating conditions with a representative core design and will be checked on a cycle-specific basis. Based on these results, the MUR-PU implementation is acceptable with respect to the final ATWS rule.

34. Fuel Evaluations

Fuel Rod Design – FSAR Section 4

Fuel rod design (FRD) analyses were performed in support of the MUR-PU of Farley Units 1 and 2 to a maximum core power level of 2831 MWt. These analyses were performed for an uprated core design where the fuel rods were irradiated at the uprated core conditions for their entire irradiation history. The current Farley Units 1 and 2 Optimized Fuel Assembly (OFA) fuel (VANTAGE+) was assumed in these uprated FRD analyses with consideration of both ZIRLO® and Optimized ZIRLO™ High Performance Fuel Cladding Material. The results of these analyses have shown that all FRD criteria were met at the MUR-PU power level. The FRD criteria evaluated included: rod internal pressure (RIP), clad oxidation, clad hydrogen pickup, clad stress, clad transient strain, clad fatigue, fuel rod axial growth, clad flattening, clad free-standing, and fuel pellet overheating (power-to-melt).

FRD analyses were performed using the NRC-approved models, methods, and criteria (Reference III.4) to guarantee that all FRD criteria are satisfied for the representative uprated neutronic models. Fuel temperatures, RIP, power-to-melt, and core stored energy were calculated to support the MUR-PU conditions to bound cycle-specific operation, using the NRC approved PAD5 fuel performance model set as defined in Reference III.4. The PAD5 fuel performance data was provided to the safety groups for use in the respective analyses. FRD analyses are performed on a cycle-specific basis considering the plant conditions of the specific cycle, as well as the fuel duty of each of the fuel regions in the core during that specific cycle. The FRD will confirm the criterion on a cycle-specific basis using the NRC approved PAD5 fuel performance model (Reference III.4) at Farley Units 1 and 2.

Core Thermal-Hydraulic – FSAR Section 4.4

Core thermal-hydraulic (T/H) analyses were performed to support the Farley MUR-PU. The analyses were based on a full core of 17x17 VANTAGE+ fuel assemblies for the uprated Farley core designs. There is no fuel design change associated with the Farley MUR-PU.

The current T/H design basis for the Farley Units 1 and 2 includes the prevention of DNB on the limiting fuel rod with a 95 percent probability at a 95 percent confidence level (95/95). The design basis is documented in FSAR Section 4.4. The MUR-PU DNB analyses are based on this licensing basis incorporating the increased core power and associated changes.

The MUR-PU DNB analyses assume a nominal core power level of 2823 MWt, which represents a 1.7 percent increase to the current nominal Farley core power. The MUR-PU

DNBR calculations that use the statistical treatment of measurement uncertainties are based on a minimum measured flow of 273,900 gpm compared to the value of 263,400 gpm used in the current DNB AOR. The higher core flow is consistent with the value in the Core Operating Limits Reports (COLRs) for the current operating cycles in the Farley units. Those DNB analyses which do not use the statistical treatment of measurement uncertainties continue to use the TDF of 258,000 gpm. The core inlet temperature used in the DNB analyses is based on the upper bound of the RCS temperature range for the MUR-PU conditions. Use of the upper-bound temperature is conservative for the DNB analyses. The DNB analyses for the MUR-PU are based on the conservative core bypass flow assumption that the RCCA are present but no other core components (thimble plugging devices, wet annular burnable absorbers, and secondary source rods) are present in the thimble tubes. The core peaking factors assumed for the MUR-PU T/H analyses remain the same as the current value in the COLRs for the current operating cycles in the Farley units.

The T/H design methods for the MUR-PU remain the same as discussed in the FSAR except for two changes:

1. The NRC-approved VIPRE-W (VIPRE) subchannel analysis code (Reference III.3) was used in place of the THINC-IV (THINC) subchannel analysis code (References III.14 and III.15) and the FACTRAN code (Reference III.9) for DNBR calculations.
2. The NRC-approved W-3 alternative correlations (ABB-NV and WLOP correlations) in Reference III.16 were used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation is not applicable.

For the implementation of the VIPRE code and the W-3 alternative DNB correlations (the ABB-NV and WLOP correlations) in the Farley DNB safety analyses, the NRC SER conditions (References III.3 and III.16) for the VIPRE code and the W-3 alternative DNB correlations were reviewed. In addition, the NRC SER conditions for the RTDP methodology were reviewed to address the implementation of the VIPRE code and the ABB-NV correlation. Compliance with these NRC SER conditions was confirmed for the DNBR analyses of the Farley units at the MUR-PU conditions as discussed below.

NRC SER Limitations and Conditions for Implementation of VIPRE and the W-3 Alternative DNB Correlations (ABB-NV and WLOP) for the Farley MUR-PU

For the DNB analyses supporting the Farley Units 1 and 2 MUR-PU, the VIPRE subchannel analysis code (Reference III.3) was used to verify that the DNB design criterion continues to be met for the VANTAGE+ fuel at MUR-PU conditions. In Reference III.3, the VIPRE code was approved for use with Westinghouse refueling methodology as a direct replacement for the THINC-IV and FACTRAN codes. Also, for the MUR-PU DNB analyses, the NRC-approved W-3 alternative correlations (ABB-NV and WLOP) in Reference III.16 are used in place of the W-3 correlation as the secondary DNB correlation for conditions where the primary DNB correlation (WRB-2) is not applicable.

For the implementation of the VIPRE code and the W-3 alternative DNB correlations in the Farley DNB limited safety analyses, the NRC SER conditions from Reference III.3 for the use of the VIPRE code and the NRC SER conditions from Reference III.16 for the use of the ABB-NV and WLOP correlations were reviewed. The DNB analyses supporting the Farley MUR-PU continue to use the RTDP methodology (Reference III.1). The NRC SER

conditions for the RTDP methodology (Reference III.1) were reviewed as well to address the change from the THINC-IV code to the VIPRE code and the use of the ABB-NV correlation with RTDP. The verification of compliance with the NRC SER conditions for these DNB-related topical reports is addressed below for the DNB analyses of the VANTAGE+ fuel in Farley Units 1 and 2 at MUR-PU conditions.

Compliance with NRC SER Conditions on the Use of VIPRE

The NRC staff reviewed Westinghouse WCAP-14565, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis," and concluded in a staff SER (Reference III.3) that the generic topical report was an acceptable reference to support plant-specific applications for use of VIPRE-01, provided four conditions identified in the SER were addressed by the licensees. These four conditions in the SER were considered in the safety analyses for Farley at the MUR-PU conditions. The VIPRE application to calculate DNBR for the VANTAGE+ fuel in Farley at MUR-PU conditions is in compliance with the four SER conditions (Reference III.3), as addressed below. The original SER conditions on the VIPRE-01 code (Reference III.26) were addressed in Reference III.3.

WCAP-14565-P-A SER Condition 1:

Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise, and other fuel-dependent parameters for a specific plant application should be justified with each submittal.

Condition 1 Response:

The WRB-2 correlation with a 95/95 correlation limit of 1.17 approved in Reference III.3 was used in the VIPRE DNBR calculations for the VANTAGE+ fuel in Farley. The ABB-NV and WLOP DNB correlations are used for the analysis of the VANTAGE+ fuel in Farley at MUR-PU conditions when the primary DNB correlation is not applicable. In Reference III.16, the ABB-NV and WLOP DNBR limits were approved for use with VIPRE. The ABB-NV and WLOP DNB correlation limits used in the MUR-PU VIPRE DNBR calculations for the VANTAGE+ fuel in Farley are consistent with the approved values in Reference III.3 for the WRB-2 correlation and Reference III.16 for the ABB-NV and WLOP DNB correlations.

There is no fuel design change associated with the Farley MUR-PU. The plant-specific hot channel factors and other fuel-dependent parameters in the DNBR calculations for the VANTAGE+ fuel in Farley at MUR-PU conditions are unchanged from the currently approved values.

WCAP-14565-P-A SER Condition 2:

Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape, and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE.

Condition 2 Response:

The core boundary conditions used in the VIPRE DNBR calculations for the VANTAGE+ fuel at MURPU conditions are all generated from NRC-approved codes and analysis

methodologies. The use of the 1.7 percent increase in the nominal core power is discussed in the safety evaluation for the MUR-PU.

The remaining reactor core boundary conditions are unchanged from the conservative values that were previously justified for the current operating license. Continued applicability of the core boundary conditions as VIPRE input is verified on a cycle-by-cycle basis using the reload methodology described in Reference III.25.

WCAP-14565-P-A SER Condition 3:

The NRC staff's generic SER for VIPRE (Reference III.26) set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2, and WRB-2M correlations. Use of other CHF correlations not currently included in VIPRE will require additional justification.

Condition 3 Response:

As discussed in response to Condition 1, the WRB-2 correlation with a 95/95 correlation limit approved in Reference III.3 was used in the VIPRE DNBR calculations for the VANTAGE+ fuel in Farley at MUR-PU conditions. The ABB-NV DNBR limit and the WLOP DNBR limit were previously approved in Reference III.16 for use with the VIPRE code.

WCAP-14565-P-A SER Condition 4:

Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to guarantee that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC staff's generic review of VIPRE (Reference III.27) did not extend to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to guarantee that conservative results are obtained.

Condition 4 Response:

For application to Farley MUR-PU safety analysis, the use of VIPRE in the post-critical heat flux region is limited to the PCT calculation for the LR transient. The calculation demonstrated that the PCT in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in Reference III.16 and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient.
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation.
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further guaranteed with the following input:

- Fuel rod input was based on the maximum fuel temperature at the given power.

- o The hot spot power factor was equal to or greater than the design linear heat rate.
- o Uncertainties were applied to the initial operating conditions in the limiting direction.

Compliance with NRC SER Conditions on the Use of RTDP

The NRC staff reviewed Westinghouse WCAP-11397, "Revised Thermal Design Procedure," and concluded in a staff SER (Reference III.1) that the generic topical report was an acceptable reference to support plant-specific applications for use of RTDP, provided seven conditions identified in the SER were addressed by the licensees. These seven conditions were considered for Farley at MUR-PU conditions. The RTDP application for the VIPRE DNB analysis of the VANTAGE+ fuel in Farley at MUR-PU conditions is in compliance with the seven SER conditions from Reference III.1, as addressed below.

WCAP-11397-P-A SER Condition 1:

Sensitivity factors used for a particular plant and their ranges of applicability should be included in the Safety Analysis Report or reload submittal.

Condition 1 Response:

Sensitivity factors were calculated using the WRB-2 and the ABB-NV DNB correlations and the VIPRE code for parameter values applicable to the VANTAGE+ fuel at Farley at MUR-PU conditions. These sensitivity factors were used to determine the RTDP design limit DNBR values for both correlations. The design limit DNBR values are included in the Farley FSAR and TS updates for the MUR-PU.

WCAP-11397-P-A SER Condition 2:

Any changes in DNB correlation, THINC-IV correlations, or parameter values listed in Table 3-1 of Reference III.1 outside of previously demonstrated acceptable ranges require re-evaluation of the sensitivity factors and of the use of Equation (2-3) of the topical report.

Condition 2 Response:

Because the VIPRE code is used to replace the THINC-IV code for the Farley MUR-PU, sensitivity factors for the RTDP methodology were calculated using the VIPRE code for parameter values applicable to the VANTAGE+ fuel at Farley at MUR-PU conditions, as discussed in the response to Condition 1 above. See the Response to SER Condition 3 for a discussion of the use of Equation (2-3) of the topical report.

WCAP-11397-P-A SER Condition 3:

If the sensitivity factors are changed as a result of correlation changes or changes in the application or use of the THINC code, then the use of an uncertainty allowance for application of Equation (2-3) must be reevaluated and the linearity assumption made to obtain Equation (2-17) of the topical report must be validated.

Condition 3 Response:

As described in Reference III.3, the VIPRE code has been demonstrated to be equivalent to the THINC code. Equation (2-3) of Reference III.1 and the linearity approximation made to obtain Equation (2-17) were confirmed to be valid for the MUR-PU for the combination of

WRB-2 correlation and the VIPRE code as well as for the combination of the ABB-NV correlation and the VIPRE code.

<u>WCAP-11397-P-A SER Condition 4:</u>

Variances and distributions for input parameters must be justified on a plant-by-plant basis until generic approval is obtained.

Condition 4 Response:

The only change to the operating parameter uncertainties for the Farley MUR-PU DNB analyses with RTDP is the reduced power calorimetric uncertainty associated with the use of a UFM to measure feedwater flow. The remaining plant operating parameter uncertainties used in the current RTDP DNB analyses are applicable to Farley at the MUR-PU conditions.

<u>WCAP-11397-P-A SER Condition 5:</u>

Nominal initial condition assumptions apply only to DNBR analyses using RTDP. Other analyses, such as overpressure calculations, require the appropriate conservative initial condition assumptions.

Condition 5 Response:

For the Farley MUR-PU, nominal initial conditions were only applied to DNBR calculations that used RTDP.

<u>WCAP-11397-P-A SER Condition 6:</u>

Nominal conditions chosen for use in analyses should bound all permitted methods of plant operation.

Condition 6 Response:

The Farley MUR-PU DNBR calculations with RTDP were based on a nominal uprated core power of 2823 MWt (1.017 * 2775 MWt). The remaining nominal conditions used in the Farley MUR-PU DNBR calculations with RTDP are unchanged from the current non-uprated values. The continued applicability of the bounding input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology described in Reference III.25.

<u>WCAP-11397-P-A SER Condition 7:</u>

The code uncertainties specified in Table 3-1 must be included in the DNBR analyses using RTDP.

Condition 7 Response:

The code uncertainties specified in Table 3-1 of Reference III.1 remain unchanged and were included in the DNBR analyses using RTDP. The THINC-IV uncertainty was applied to VIPRE, based on the equivalence of the VIPRE model approved in Reference III.3 to THINC-IV.

Compliance with NRC SER Conditions on the Use of the W-3 Alternative DNB Correlations (ABB-NV and WLOP)

The NRC staff reviewed Westinghouse WCAP-14565-P-A, Addendum 2, "Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV

Correlation WLOP for PWR Low Pressure Applications," (Reference III.16) and concluded in a staff SER (Reference III.16) that the generic topical report was acceptable for licensing applications, subject to the four limitations and conditions identified in the SER being addressed by the licensees. These four limitations and conditions in the SER were considered in the safety analyses for Farley at the MUR-PU conditions. The application of the ABB-NV and WLOP correlations to calculate DNBR for the VANTAGE+ fuel in Farley at MUR-PU conditions is in compliance with the four limitations and conditions from Reference III.16, as addressed below.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 1:

The applicable range of the ABB-NV and WLOP correlations are presented in Tables 1 and 2, respectively, of this SE.

Condition 1 Response:

For the DNB analyses at MUR-PU conditions that were based on the ABB-NV and WLOP correlations, the results were confirmed to be within the parameter ranges of the DNB correlations as specified in Tables 1 and 2, respectively (Reference III.16).

WCAP-14565-P-A, Addendum 2-P-A SER Condition 2:

The ABB-NV correlation and the WLOP correlation must use the same Fc factor for power shape correction as used in the primary DNB correlation for a specific fuel design.

Condition 2 Response:

For the DNB analyses at MUR-PU conditions that were based on the ABB-NV and WLOP correlations, the Fc factor for power shape correction that was applied was the same as the power shape correction used for the WRB-2 correlation, which is the primary DNB correlation for the VANTAGE+ fuel in Farley.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 3:

Selection of the appropriate DNB correlation, DNBR limit, engineering hot channel factors for enthalpy rise, and other fuel-dependent parameters will be justified for each application of each correlation on a plant specific basis.

Condition 3 Response:

The ABB-NV and WLOP DNB correlations are used for analysis of the VANTAGE+ fuel in Farley at MUR-PU conditions when the primary DNB correlation is not applicable. In Reference III.16, the current ABB-NV and WLOP DNBR limits were approved for use with VIPRE. The 95/95 ABB-NV and 95/95 WLOP DNB correlation limit used in the MUR-PU VIPRE DNBR calculations for the VANTAGE+ fuel in Farley are consistent with the approved values in Reference III.16. There is no fuel design change associated with the Farley MUR-PU. The plant-specific hot channel factors and other fuel-dependent parameters in the DNBR calculations for the VANTAGE+ fuel in Farley at MUR-PU conditions are unchanged from the currently approved values.

WCAP-14565-P-A, Addendum 2-P-A SER Condition 4:

The ABB-NV correlation for Westinghouse PWR applications and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 (VIPRE) code

since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications.

Condition 4 Response:

The Westinghouse version of the VIPRE-01 code subchannel analysis code (Reference III.3), which has been qualified and approved with the ABB-NV and WLOP correlations, was implemented for DNB analyses of the VANTAGE+ fuel in Farley at MUR-PU conditions where the primary DNB correlation (WRB-2) was not applicable.

Subchannel Analysis Code

In Reference III.3, the VIPRE code was demonstrated to be equivalent to THINC-IV and FACTRAN, and was approved by the NRC for use with the Westinghouse refueling methodology as a direct replacement for the THINC-IV and FACTRAN codes. The use of VIPRE for the MUR-PU analysis is in full compliance with the conditions specified in the NRC SER for Reference III.3, as discussed above.

The change from THINC to VIPRE is necessary to implement the NRC-approved W-3 alternative DNB correlations from Reference 5 (the ABB-NV and WLOP correlations) for use in the MUR-PU analyses as supplemental DNB correlations. The SER for Reference III.16 requires that the ABB-NV correlation for Westinghouse PWR application and the WLOP correlation must be used in conjunction with the Westinghouse version of the VIPRE-01 code since the correlations were justified and developed based on VIPRE and the associated VIPRE modeling specifications. In addition, use of VIPRE is needed to generate input for the implementation of the mechanistic DNB propagation methodology (Reference III.17). The mechanistic DNB propagation methodology from Reference III.17 is consistent with that defined in Reference III.4. There were no deviations from the NRC-approved methodology that requires no rods burst and no rods strain enough to block flow in a channel.

DNBR calculations were performed with the VIPRE code for the DNB-limited FSAR Chapter 15 events described above (Section III.1) that are currently analyzed with the THINC subchannel analysis code. The DNBR calculations performed with the VIPRE code address the increased nominal power and the change in power measurement uncertainty associated with the MUR-PU.

DNB Correlations and Limits

Consistent with the FSAR, the primary DNB correlation for the analysis of the VANTAGE+ fuel at MUR-PU conditions continues to be the WRB-2 DNB correlation (Reference III.18). The DNB correlation limit is determined on the basis of the scatter in the measured-to-predicted results from the subchannel analysis of the CHF test data. For the current Farley DNB analysis, the W-3 DNB correlation (Reference III.19) is used to supplement the primary DNB correlation where the primary DNB correlation is not applicable as discussed in Subsection 4.4.2.3.1 of the FSAR. The W-3 correlation is insufficient to provide the DNBR margin necessary to support future reload cores at the MUR-PU conditions. For the MUR-PU DNB analyses, the NRC-approved W-3 alternative DNB correlations from Reference III.16, are used as the secondary DNB correlations.

DNB Methodology

The DNB analyses of the VANTAGE+ fuel in Farley Units 1 and 2 at MUR-PU conditions continue to be based on the RTDP (Reference III.1). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain the overall DNB uncertainty factors. Proprietary DNBR sensitivity factors, which are used to develop DNB uncertainty factors, are calculated over ranges of conditions that bound the events for which RTDP methodology is applied. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is a 95 percent probability with a 95 percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, or transient conditions arising from faults of moderate frequency.

The current RTDP design limit DNBR values for the VANTAGE+ fuel in the Farley units were based on the THINC-IV code and the WRB-2 correlation. The continued applicability of these RTDP design limit DNBR values for the VANTAGE+ fuel in the Farley units at MUR-PU conditions was confirmed for the use of the VIPRE code in place of the THINC-IV code. The reduced power measurement uncertainty provides a benefit in the calculation of the RTDP design limit DNBR values. The proprietary DNBR sensitivity factors required to develop the WRB-2 DNB uncertainty factors for the MUR-PU were based on the VIPRE code. Based on the applicable parameter uncertainties and the VIPRE-based DNBR sensitivity factors, the current RTDP design limit DNBR values were shown to remain applicable for the DNBR analyses using VIPRE and the WRB-2 correlation at the MUR-PU conditions.

The ABB-NV correlation is applicable for the DNB analysis of the portion of the fuel rod below the first mixing vane grid. An RTDP design limit DNBR was developed for the use of the ABB-NV correlation at the MUR-PU conditions. The proprietary DNBR sensitivity factors required to develop the ABB-NV DNB uncertainty factors for the MUR-PU were based on the VIPRE code. Based on the applicable parameter uncertainties and the VIPRE-based DNBR sensitivity factors, an RTDP design limit DNBR value was established for the DNBR analyses using VIPRE and the ABB-NV correlation at the MUR-PU conditions. In addition to the preceding considerations for uncertainties, DNBR margin is retained by performing the safety analyses to DNBR limits higher than the RTDP design limit DNBR values. Sufficient DNBR margin is conservatively maintained in the safety analysis DNBR limits to offset the rod bow DNBR penalty and to provide flexibility in design and operation of the plant.

For the MUR-PU DNB analyses, the STDP continues to be used for those DNB analyses where RTDP is not applicable. For the STDP, the initial condition uncertainties are accounted for deterministically by applying the uncertainties to the nominal conditions. The DNBR limit for STDP is the appropriate DNB correlation limit with consideration for applicable DNBR penalties.

Effects of Fuel Rod Bow on DNBR

Rod bow can occur in the spans between the grids, reducing the spacing between adjacent fuel rods and reducing the margin to DNB. Rod bow must be accounted for in the DNB safety analysis of Condition I and Condition II events. For MUR-PU conditions, the rod bow

DNBR penalty continues to be based on the NRC-approved methodology in References III.20, III.21, and III.22. For the VANTAGE+ fuel, the appropriate rod bow DNBR penalty depends on the grid span. The maximum DNBR penalty for rod bow in the mixing vane grid region of the VANTAGE+ fuel is based on the WRB-2 correlation. The maximum DNBR penalty for rod bow in the region below the first mixing vane grid is based on the ABB-NV correlation. For the grid spans that contain an intermediate flow mixer (IFM) grid, additional restraint is provided by the IFM grids such that the grid-to-grid spacing is reduced to approximately 10 inches. No rod bow DNBR penalty is required in the 10-inch spans in the VANTAGE+ safety analyses.

Acceptance Criterion

The T/H design basis for the MUR-PU remains the same as discussed in the FSAR. The DNB design basis for the MUR-PU DNB analysis is that there will be at least a 95 percent probability at 95 percent confidence level that DNB will not occur on the limiting fuel rods during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events). Analytical assurance that the DNB criterion is met is provided by showing that the VIPRE-calculated DNBR is higher than the appropriate 95/95 DNBR limit for the DNB methodology and DNB correlation used in the analysis and that the VIPRE results are within the parameter ranges of the DNB correlation.

Description of Analyses

Core Thermal Limits

The core thermal limits are required for the generation of the overtemperature delta T (OTΔT) and overpower delta T (OPΔT) reactor trip setpoints. The core thermal limits define the loci of points of thermal power, primary system pressure, and coolant inlet temperature that satisfy the following criteria:

- The minimum DNBR is not less than the SAL DNBR.
- The hot channel exit quality is not greater than the upper limit of the quality range of the DNB correlation (adjusted for the analysis-specific quality uncertainty).
- Vessel $T_{hot} < T_{sat}$ to guarantee that the difference between T_{hot} and T_{cold} remains proportional to power.

To support operation at MUR-PU conditions, new core thermal limits were generated for the VANTAGE+ fuel in the Farley units. The DNB-limited portion of the MUR-PU core thermal limits was generated with the VIPRE code using the WRB-2 DNB correlation and the RTDP methodology.

DTDI Limits

The delta-T delta-I (DTDI) limits are used to reduce the core DNB limit lines to account for the effect of adverse axial power distributions that are more limiting for DNB than the axial power shape used to generate the core thermal limits. New DTDI limits were generated for the VANTAGE+ fuel in the Farley units to address the MUR-PU conditions. The MUR-PU DTDI limits were generated with the VIPRE code using the RTDP methodology. For the DNB analysis of axial power distributions that were limiting in the fuel region above the first mixing vane grid, the WRB-2 DNB correlation was used. For the DNB analysis of axial

power distributions that were limiting in the fuel region below the first mixing vane grid, the ABB-NV DNB correlation was used. The MUR-PU DTDI limits were used to define the f(ΔI) reset function in the OTΔT reactor trip function such that the DNB design criterion is met for accidents that are terminated by the OTΔT reactor trip function.

Loss of Flow

As noted above in section III.1, the loss-of-flow accident was analyzed for MUR-PU conditions. The DNBR calculations for the loss-of-flow accident at MUR-PU conditions were performed using the VIPRE code to replace THINC-IV and FACTRAN. The DNBR calculations were based on the WRB-2 DNB correlation and the RTDP methodology. The effect of fuel temperatures was included in the analysis of this event. Three cases (partial loss of flow, complete loss of flow, and frequency decay complete loss of flow) were analyzed to guarantee the limiting scenario was identified. The frequency decay complete loss of flow case results in the lowest minimum DNBR. The minimum DNBR values calculated for the three cases are greater than the WRB-2 RTDP SAL DNBR, thereby demonstrating compliance with the DNB design criterion for this event. DNB analysis was also performed to confirm that the DNB criterion was met for low flow conditions supporting the P-8 setpoint. This analysis uses STDP and the WRB-2 correlation. The calculated minimum DNBR is above the correlation DNBR limit with sufficient margin to account for applicable DNBR penalties.

Locked Rotor (Rods in DNB)

As noted above in section III.1, the LR accident was analyzed for MUR-PU conditions. The LR accident is classified as a Condition IV event. DNBR calculations are performed to quantify the inventory of rods that would undergo DNB and be conservatively presumed to fail. The DNBR calculations for the LR rods-in-DNB event at MUR-PU conditions were performed using the VIPRE code to replace THINC-IV and FACTRAN. The DNBR calculations were based on the WRB-2 DNB correlation and the RTDP methodology. The effect of fuel temperatures was included in the transient VIPRE model analysis of this event.

The LR PCT analysis is performed using STDP and the VIPRE code. The PCT analysis for the Farley MUR satisfied the acceptance criterion, confirming that the fuel melt limit for Optimized ZIRLO™ high performance fuel cladding material is met. A mechanistic DNB propagation analysis was performed as part of the MUR-PU. The new analysis shows that no rods burst and no rods strain enough to block flow in a channel.

Hot Zero-Power Steam Line Break Accident

As noted above in section III.1, the hot zero-power steam line break (HZP SLB) event was confirmed for the MUR-PU. The NRC-approved Westinghouse analysis method in Reference III.23 was used for analyzing the HZP SLB accident. DNBR calculations for the HZP SLB event were performed at MUR-PU conditions using the VIPRE code, the WLOP correlation, and the STDP methodology. The WLOP correlation was used for this application because the system pressure was less than the low-pressure limit of applicability for the primary WRB-2 DNB correlation. The STDP methodology was used because the event is initiated from HZP conditions. Conservative accident-specific axial and radial power distributions were applied. For this STDP application, the applicable DNBR limit is the 95/95 correlation limit for the WLOP correlation. The results of the VIPRE calculations at MUR-PU

conditions show that the minimum DNBR for the HZP SLB event is greater than the WLOP correlation DNBR limit. Therefore, the DNB design basis is met for the HZP SLB event at MUR-PU conditions.

Hot Full Power Steam Line Break Accident

As noted above in section III.1, the HFP SLB event was analyzed for MUR-PU conditions. DNBR calculations for the HFP SLB accident at MUR-PU conditions were performed using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology. The DNBR results at MUR- PU conditions show that the minimum DNBR is greater than the WRB-2 RTDP SAL DNBR, thereby demonstrating that the DNB design basis is met.

RCCA Drop/Misoperation

The NRC-approved Westinghouse analysis methods in Reference III.24 continue to be used for analyzing the RCCA drop event at MUR-PU conditions. dropped rod limit lines (DRLL) were generated to define the loci of points that result in a VIPRE-calculated minimum DNBR equal to the WRB-2 RTDP SAL DNBR for a wide range of core conditions (inlet temperature, power, and pressure). The DRLL are used to verify that the DNB design basis is met each cycle for the RCCA drop event at MUR-PU conditions.

The maximum allowable FN ΔH limit for RCCA misalignment was determined using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology. This is the value of FN ΔH at normal operating conditions that results in a minimum DNBR equal to the WRB-2 RTDP SAL DNBR. The limits provided for the RCCA drop and RCCA misalignment events are used to confirm that the DNB design basis is met for Farley reload cores operating at MUR-PU conditions.

Uncontrolled Rod Cluster Control Assembly Withdrawal from Subcritical

As noted above in section III.1, the statepoints for this zero-power event are unaffected by the 1.7 percent core power uprate. The limiting heat flux statepoints are defined as a fraction of the nominal heat flux. DNBR calculations for the uncontrolled RCCA RWFS event were performed at MUR-PU conditions to incorporate the VIPRE code and the ABB-NV correlation. The DNBR calculations for this event were based on the STDP methodology, since the event is initiated from HZP conditions. Conservative accident-specific axial and radial power distributions were used in the DNB analysis. Two DNBR calculations were required for this event. The ABB-NV correlation was applied in the fuel region below the first mixing vane grid. The WRB-2 correlation was applied in the fuel region above the first mixing vane grid. For this STDP application, the DNBR limits applied are the 95/95 correlation limits for the ABB-NV and WRB-2 correlations. The results of the VIPRE calculations with the 1.7 percent increase in the nominal heat flux show that the minimum DNBR values for the RWFS event, including the appropriate DNBR penalties for this event, are greater than the correlation DNBR limits for the ABB-NV and WRB-2 correlations. Therefore, the DNB design basis is met for the RWFS event at MUR-PU conditions.

Rod Withdrawal at Power

As noted above in section III.1, the RWAP event was analyzed for MUR-PU conditions. DNBR calculations for the RWAP accident at MUR-PU conditions were performed using the VIPRE code, the WRB-2 DNB correlation, and the RTDP methodology. The RWAP DNBR

results at MUR-PU conditions did not meet the WRB-2 RTDP SAL DNBR, thereby incurring a small DNBR penalty which is offset by DNBR margin. The design basis is met for this event.

Analyses described in the previous sections show that the DNB design basis is met for Farley Units 1 and 2 at MUR-PU conditions. Cycle-specific evaluations to confirm that the DNB design basis is met for each reload at MUR-PU conditions will be performed in accordance with Reference III.25.

References for Section III.1

III.1 Westinghouse Report, WCAP-11397-P-A, Revision 0, "Revised Thermal Design Procedure," April 1989 (Westinghouse Proprietary) and WCAP-11397-A, Revision 0 (Non-proprietary).

III.2 Westinghouse Report, WCAP-7907-P-A, Revision 0, "LOFTRAN Code Description," April 1984 (Westinghouse Proprietary) and WCAP-7907-A, Revision 0 (Non-proprietary).

III.3 Westinghouse Report, WCAP-14565-P-A, Revision 0, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999 (Westinghouse Proprietary) and WCAP-15306-NP-A, Revision 0 (Non-proprietary). (ML99316009)

III.4 Westinghouse Report, WCAP-17642-P-A, Revision 1, "Westinghouse Performance Analysis and Design Model (PAD5)," November 2017 (Westinghouse Proprietary) and WCAP-17642-NP-A, Revision 1 (Non-proprietary).

III.5 Westinghouse Letter from D. E. McKinnon (Westinghouse) to L. K. Mathews (Southern Company), 90AP*-G-0041, "Joseph M. Farley Nuclear Plant Units 1 and 2 Operating Procedure for Mode 4/5 Boron Dilution," July 6, 1990 (Non-proprietary).

III.6 Westinghouse Report, WCAP-14882-P-A, Revision 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999 (Westinghouse Proprietary) and WCAP-15234-A, Revision 0 (Non-proprietary).

III.7 Westinghouse Report, WCAP-14882-S1-P-A, Revision 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactors Non-LOCA Safety Analyses, Supplement 1 – Thick Metal Mass Heat Transfer Model and NOTRUMP-Based Steam Generator Mass Calculation Method," October 2005 (Westinghouse Proprietary) and WCAP-15234-S1-A, Revision 0 (Non-proprietary).

III.8 Westinghouse Report, WCAP-7979-P-A, Revision 0, "TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code," January 1975 (Westinghouse Proprietary) and WCAP-8028-A, Revision 0 (Non-proprietary).

III.9 Westinghouse Report, WCAP-7908-A, Revision 0, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989 (Non-proprietary).

III.10 Westinghouse Report, WCAP-7588, Revision 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975 (Non-proprietary).

III.11 10CFR50.62, "Requirements for Reduction of Risk from ATWS Events for Light Water-Cooled Nuclear Power Plants."

III.12 Westinghouse Letter, NS-TMA-2182, "Anticipated Transients Without Scram for Westinghouse Plants," December 1979.

III.13 ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers.

III.14 Westinghouse Report, WCAP-7956-A, "THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," February 1989. (Westinghouse Proprietary)

III.15 Westinghouse Report, WCAP-12330-A, "Improved THINC-IV Modeling for PWR Core Design," September 1991. (Westinghouse Proprietary)

III.16 Westinghouse Report, WCAP-14565-P-A, Addendum 2-P-A, "Addendum 2 to WCAP-14565-PA, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," April 2008. (Westinghouse Proprietary)

III.17 Westinghouse Report, WCAP-8963-P-A, Addendum 1-A, Revision 1-A, "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis (Departure from Nucleate Boiling Mechanistic Propagation Methodology)," June 2006. (Westinghouse Proprietary)

III.18 Westinghouse Report, WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly, September 1985. (Westinghouse Proprietary)

III.19 L. S. Tong, AEC Critical Review Series, "Boiling Crisis and Critical Heat Flux," TID-25887, August 1972.

III.20 Westinghouse Report, WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979. (Westinghouse Proprietary)

III.21 Letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NS-EPR-2515, October 9, 1981; and Letter from Rahe, E. P., Jr. (Westinghouse) to Miller, J. R. (NRC), "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1," NSEPR- 2572, March 16, 1982.

III.22 Letter from Berlinger, C. (NRC) to Rahe, E. P., Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty, June 18, 1986.

III.23 Westinghouse Report, WCAP-9226-P-A, Revision 1, "Reactor Core Response to Excessive Secondary Steam Releases," Scherder, W. J. (Editor), et al., February 1998. (Westinghouse Proprietary)

III.24 Westinghouse Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990. (Westinghouse Proprietary)

III.25 Westinghouse Report, WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985. (Westinghouse Proprietary)

III.26 Safety Evaluation for Amendment No. 147 to NPF-2 and Amendment No. 138 to NPF-8, Dated December 29, 1999 (Replacement Steam Generators).

III.27 Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA Executive Committee), "Acceptance for Referencing of Licensing Topical Report, EPRI-NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,' Volumes 1, 2, 3 and 4," May 1, 1986.

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:

Farley Units 1 and 2 have no reload analyses that require re-evaluation for the MUR-PU. Various reload analyses are performed for each fuel cycle in accordance with normal cycle design practice and included in the COLR per Farley TS 5.6.5.

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

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

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

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

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

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

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

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

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

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

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

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

RESPONSE:

All Farley Unit 1 and 2 analyses of record for the FSAR Chapter 15 accident analyses as well as other analyses that support licensing of the plant support the MUR-PU as described in Section II.1.

All other analyses that are included in Section III.1 support the MUR-PU and PAD5 implementation. These analyses will be incorporated as the AOR for their respective FSAR sections under 10 CFR 50.59 for implementation of the MUR-PU.

References for Section III:

See the individual reference listing for Section III.1.

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:

NSSS Design Parameters

Table IV-1 presents a summary of the NSSS critical thermal design parameters, for both current operating conditions and the proposed MUR-PU. MUR-PU data is shown for maximum analytical reactor thermal power of 2831 MWt (102 percent of 2775 MWt). Licensed thermal power will be 2821 MWt. No SG tube plugging (SGTP) is assumed for these parameters.

All the plant components described in this section do not assume any uncertainty on power level. The increase in power and corresponding change in design temperatures was evaluated for any impact on the applicable design criteria. The MUR-PU analyzed power level of 2831 MWt reactor power/2841 MWt NSSS power was assumed as a bounding condition, unless otherwise noted.

Table IV-1: Critical Thermal Design Parameters

Thermal Design Parameters	Current		MUR Power Uprate	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
Primary Side				
Reactor Power – Analyzed (MWt)	2775*	2775*	2831	2831
Reactor Power – Licensed (MWt)	2775	2775	2821	2821
NSSS Power – Analyzed (MWt)	2785	2785	2841	2841
Reactor Coolant Thermal Design Flow (gpm)	86,000	86,000	86,000	86,000
Reactor Coolant Flow (E+06 lbm/hr)	98.1	99.5	98.3	98.3
Mechanical Design Flow (gpm)	101,800	101,800	101,800	101,800
Reactor Coolant Pressure (psia)	2250	2250	2250	2250
Reactor Coolant Vessel Outlet (T_{hot}) (°F)	613.3	603.8	614.0	604.5
Reactor Coolant Vessel Average (T_{avg}) (°F)	577.2	567.2	577.2	567.2
Reactor Coolant Vessel/Core Inlet (T_{cold}) (°F)	541.1	530.6	540.5	529.9

Thermal Design Parameters	Current		MUR Power Uprate	
	High T_{avg}	Low T_{avg}	High T_{avg}	Low T_{avg}
Secondary Side				
Steam Generator Outlet Temperature (°F)	515.5	504.6	514.4	503.4
Steam Generator Outlet Pressure (psia)	781	709	773	702
Steam Flow (E+06 lbm/hr)	12.24	12.22	12.54	12.50
Feedwater Temperature (°F)	443.4	443.4	446.0	446.0
* For safety analyses, 2-percent uncertainty was typically added on to reactor power for the analyzed condition				

The current T_{avg} temperature range was maintained for the Farley Units 1 and 2 MUR-PU, namely from 567.2°F to 577.2°F. The MUR-PU does result in a small increase of 0.7°F in the bounding vessel outlet temperature (T_{hot}) (i.e., high T_{avg}/ maximum T_{hot} case) and decrease of 0.7°F in the bounding vessel inlet temperature (T_{cold}) (i.e., low T_{avg}/ minimum T_{cold} case), as shown in the parameters in Table IV-1. However, it is recognized that these limiting temperature conditions are not achievable in the plant due to the overly conservative assumptions used in developing the NSSS design parameters, namely the conservatively low TDF and the maximum and minimum T_{avg} range limits of 577.2°F and 567.2°F. While the small changes in the T_{hot} and T_{cold} limits specified in Table IV-1 were considered in most of the NSSS structural analyses, in some cases this operational limitation was recognized in order to simplify the evaluation. This will be noted within the appropriate section if this was the case.

<u>NSSS Design Transients</u>

The NSSS design transients were also reviewed for any impact due to the MUR-PU. The NSSS design transients' evaluation is based on the NSSS design parameters discussed in the previous section. The evaluation reviewed the existing design transients from the AOR for their continued applicability at the MUR-PU conditions.

The Farley Units 1 and 2 NSSS design transients from the AOR are primarily based on bounding plant operating conditions that conservatively enveloped the Farley Units 1 and 2 design parameters. The bounding operating conditions used in the current design transients were compared to the Farley Units 1 and 2 MUR-PU conditions to guarantee that the existing design transients remain conservative for the MUR-PU program.

The maximum full power T_{hot} increased from 613.3°F to 614.0°F for the MUR-PU Program. However, an operational limit specifies that the maximum T_{hot} is limited to 613.3°F for the MUR-PU. Therefore, this limitation maintains the minimum T_{hot} for the MUR-PU within the existing bounds of the AOR. Furthermore, the T_{hot} values for the MUR-PU fall well within the limits of the bounding plant parameters that were used as the basis for the majority of the Farley Units 1 and 2 NSSS design transients. A higher full power T_{hot} is conservative for the NSSS design transients; hence, the T_{hot} responses specified by the current design transients remain conservative.

The minimum full power T_{cold} decreased from 530.6°F to 529.9°F for the MUR-PU program. However, an operational limit specifies that the minimum T_{cold} is limited to 530.6°F for the MUR-

PU. Therefore, this limitation maintains the minimum T_{cold} for the MUR-PU within the existing bounds of the Stretch Power Uprate and Replacement Steam Generator (RSG) programs. Furthermore, the T_{cold} values for the MUR-PU, even without the operational limitations, fall well within the limits of the bounding plant parameters that were used as the basis for the majority of the Farley Units 1 and 2 NSSS design transients. A lower T_{cold} is conservative for the NSSS design transients; hence, the T_{cold} responses specified by the current design transients remain conservative.

The RCS pressure, TDF, and mechanical design flow (MDF) values are identical for the MUR-PU compared to the AOR. The NSSS power is about 2 percent higher for the MUR-PU than the AOR. The full power primary side temperatures listed in Table IV-1 are based on the NSSS power of 2841 MWt. As such, the effect of the increased MUR-PU NSSS power level is reflected in the revised primary side RCS temperatures for the MUR-PU program, which were shown to be bounded by those used in the current design transients. Therefore, no further evaluation of the NSSS power increase is required for the design transients.

The MUR-PU covers up to 15 percent average SGTP and 20 percent peak SGTP in any SG. The SG pressures and corresponding temperatures are up to 8 psi lower at MUR-PU conditions than for the RSG conditions. However, the existing operational limit of ≥ 690 psia for 15 percent SGTP in the AOR also applies for the MUR-PU since the plant cannot run at full power at conservative steam pressures below this value. Therefore, the minimum allowed SG pressure (and corresponding temperature) for the MUR-PU is bounded by the current minimum SG pressure for the AOR. Furthermore, the bounding plant conditions that are the basis of the Farley Units 1 and 2 design transients used a significantly lower full power steam pressure which envelopes all full power steam pressures listed in Table IV-1.

As shown in Table IV-1, the maximum full power feedwater temperature (T_{feed}) for the MUR-PU program is 2.6°F higher than the maximum full power T_{feed} for the current AOR. For the secondary side SG transients, a higher T_{feed} is conservative. However, the secondary side parameters have a minor effect on the primary side transient responses. Therefore, the primary side transient responses developed based on a lower T_{feed} remain valid for the increased feedwater temperature at MUR-PU conditions.

As a result of the increased full power T_{feed}, the T_{feed} response to several transients was revised for use in the secondary side SG analyses. Additionally, the full power feedwater flow (equal to steam flow) for the MUR-PU is slightly higher than that of the current design transients (see Table IV-1). Although both the steam flow and feedwater flow increased for the MUR-PU, the design transient curves specify these flows as an FON. Therefore, the majority of the curves remain valid, but the initial condition applied to the curves should be increased to reflect the nominal steam flow and feedwater flow at MUR-PU conditions listed in Table IV-1. Steam flow and feedwater flow curves that were initiated from 102 percent of nominal have been revised to initiate from 100 percent of the MUR-PU flow rate since the 2 percent allowance is no longer applicable to the MUR-PU.

Conclusion

The existing NSSS primary side and pressurizer design transients remain valid for the Farley Units 1 and 2 MUR-PU program. For the secondary side, revised T_{feed}, steam flow and

feedwater flow curves were developed to account for the MUR-PU conditions. Any new transient curves were utilized in the evaluation of the SGs for the MUR-PU.

The MUR-PU does not result in any new transients that must be considered, nor does it affect the number of transient occurrences currently specified. Therefore, the transients and their associated frequencies of occurrences listed in FSAR Table 5.2-2 remain valid for the MUR-PU.

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

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

RESPONSE:

The revised operating conditions were reviewed for impact on the existing design basis analyses for the RV. No changes in RCS design or operating pressure were made as part of the MUR-PU. The effects of operating temperature changes (RV outlet, RV inlet) are within design limits. The current NSSS design transients remain applicable to the RV, and no additional transients have been proposed. In addition, the LOCA hydraulic forces for the current analysis remain valid. Therefore, the current faulted condition blowdown LOCA plus safe shutdown earthquake seismic loads considered in the structural analysis of the RV remain bounding.

As a result, there are no changes to the current maximum stress intensities, the maximum ranges of stress intensity, or the maximum cumulative fatigue usage factors for the RV assemblies, nozzles and supports, as well as the replacement RV closure heads. The ASME code of record for these components listed in Section IV.1.D remains unchanged for plant operation at the uprated power conditions.

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

RESPONSE:

Evaluations were performed to assess the impact of the MUR-PU on the following areas within the core support structures and vessel internals.

IV.1.A.ii.a Thermal-Hydraulic Systems Evaluation

The hydraulic behavior of the coolant flow and its effect on the reactor internals system was evaluated. The parameters which could cause some effect are the T_{hot} and T_{cold} temperatures. As shown in Table IV-1, the maximum T_{hot} increases by 0.7°F and the minimum T_{cold} decreases by 0.7°F due to the increase in power. It was determined that these temperature differences are within design values and have an insignificant impact on the following areas due to the small amount of change.

- Design core bypass flow of 7.1 percent
- Reactor internals pressure drop
- Hydraulic lift forces, and the hold-down capability of the internals
- Upper head fluid temperature
- Baffle joint momentum flux and fuel rod stability

IV.1.A.ii.b Flow-Induced Vibration

The RV internals are subjected to vibrations induced by flow turbulences and vortex shedding. High frequency acoustic sources from RCPs and low frequency acoustic sources from loop oscillations can induce vibrations in the internals during steady state operation conditions.

The flow-induced vibration (FIV) phenomena of the reactor internals can potentially be influenced by changes in the T_{cold}, T_{hot}, and hydraulic design parameters such as TDFs or MDFs. As shown in Table IV-1, the lowest T_{cold} decreases by 0.7°F and the highest T_{hot} increases by 0.7°F. This temperature change results in a change in water density that has a negligible impact on the vibratory response of the reactor internals.

The Farley Units 1 and 2 current design basis and the MUR-PU parameters are essentially the same; therefore, it can be concluded that there is no significant impact on the performance of the reactor internals with regard to FIV.

IV.1.A.ii.c Reactor Internals Heat Generation Rates

The presence of radiation-induced heat generation in reactor internals components, in conjunction with the various reactor coolant fluid temperatures, results in thermal gradients within and between the components. These thermal gradients cause thermal stress and thermal growth, which must be considered in the design and analysis of the various components. The primary design considerations are to guarantee that thermal growth is consistent with the functional requirements of the components, and to guarantee that the applicable ASME Code requirements are satisfied as part of the component evaluation.

The reactor internals components subjected to significant radiation-induced heat generation are the upper core plates (UCPs) and lower core plates (LCPs), core baffle plates, former plates, core barrel, baffle-former bolts, and barrel-former bolts. Due to the relatively low HGRs in the lower core support and the neutron pads, these components experience little, if any, temperature rise relative to the surrounding reactor coolant.

For the radial reactor internals components, long-term HGRs were analyzed, since it was anticipated that the gamma heating rates would be bounded by the corresponding design gamma heating rates reported in Reference IV.1.A.1. The Reference IV.1.A.1 gamma heating rates utilize conservative design parameters meant to maximize the exposure to radial reactor components, such as the use of out-in loading patterns, whereas, Farley Units 1 and 2 have transitioned in the past to low-leakage loading patterns. Since the long-term radial case of Reference IV.1.A.1 was shown to be bounding, the short-term radial case of Reference IV.1.A.1 would also remain bounding and, therefore, was not quantitatively evaluated. The MUR-PU long-term HGR evaluation of the core baffle plates and core barrel for each unit was based on the average spatial distribution of recent fuel cycles, in conjunction with the uprated core power level (2831 MWt) and associated reactor operating conditions.

Design basis HGRs applicable to the radial internals were obtained from Appendices H and I of Reference IV.1.A.1. The core power distributions upon which those calculations were derived from statistical studies of 25 independent fuel cycles from 11 three-loop reactors. These power distributions represented an upper tolerance limit for beginning of cycle and end of cycle (EOC) power in the peripheral fuel assemblies, based on a 95-percent probability with a 95-percent

confidence level. Most of the evaluated fuel cycles were based on an out-in fuel loading strategy (fresh fuel on the periphery) which, when combined with the statistical processing of the data, resulted in a design basis core power distribution that tended to be biased high on the periphery. This high bias on the core periphery was used to guarantee conservative, but realistic, design calculations for the critical baffle-barrel region of the reactor internals and explains why the Reference IV.1.A.1 radial component heating rate results were bounding for the Farley Units 1 and 2 MUR-PU.

For the UCPs and LCPs, long-term HGRs were analyzed since it was anticipated that the gamma heating rates would be bounded by the corresponding design gamma heating rates applicable to Westinghouse 3-loop reactor designs. Note that the heating rates were determined for both long- and short-term conditions using more conservative assumptions than were used in Reference IV.1.A.1. More specifically, the heating rates utilized conservative design parameters such as a flat axial power distribution to maximize the exposure to axial reactor components. Since the long-term case applicable to Westinghouse 3-loop reactor designs was shown to be bounding, the short-term cases would also remain bounding and, therefore, were not quantitatively evaluated.

IV.1.A.ii.d Reactor Internals Core Support Structures

Evaluations were performed to demonstrate that the structural integrity of the reactor internal components is not adversely affected by the MUR-PU conditions. The presence of heat generated in the reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between the components. These thermal gradients result in thermal stresses and thermal growth that must be accounted for in the design and analysis of various reactor internal components.

The reactor internals and core support structure components subjected to heat generation effects (either directly or indirectly) are the UCP, the LCP, and the baffle-barrel region. An evaluation concluded that the UCP, LCP, and baffle-barrel region AOR all remained valid under the MUR conditions along with the change due to heat generation effects. The stresses and fatigue usage factors are within an acceptable range with the MUR-PU.

IV.1.A.iii Control rod drive mechanisms

RESPONSE:

The MUR-PU program affects only small changes to the plant operating parameters, namely T_{hot} and T_{cold}, and does not impact the physical plant layout. It is also concluded that the design transient definitions and frequency of occurrences applicable to current power conditions remain applicable for MUR-PU conditions. Therefore, the seismic response of the NSSS is not significantly impacted by the MUR-PU.

The impact of the MUR-PU on LOCA hydraulic loading was also evaluated and determined to be insignificant. Therefore, the response of the reactor assembly to this LOCA hydraulic loading would not be affected, and the LOCA stresses calculated in the CRDM AOR remain valid for MUR-PU conditions.

Since all relevant plant parameters pertaining to the structural evaluation of the CRDMs are not significantly impacted by MUR-PU conditions, the CRDM design report remains applicable without modification.

IV.1.A.iv Nuclear steam supply system (NSSS) piping, pipe supports, branch nozzles

RESPONSE:

For the MUR-PU, there were no physical plant modifications to the RCS piping system that impacted the piping stress analyses. The changes were limited to operational changes, such as those associated with the temperature, pressure, and flow characteristics of the reactor coolant. The evaluation of the RCS piping analyses reviewed the inputs used in the piping analyses that are potentially impacted by operational changes, specifically the NSSS design parameters in Table IV-1, NSSS design transients, LOCA hydraulic forces, or LOCA RPV motions.

The evaluation determined that the changes associated with the MUR-PU either had no impact or did not have a significant impact on the RCS piping analyses, including the branch nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge). Therefore, the existing RCS piping analyses for the Farley Units 1 and 2 AOR remain applicable and valid for the MUR-PU Program.

Since there was no impact to the RCS piping analyses as a result of the MUR-PU, there were no impacts to the following outputs from the piping analyses that are used as inputs in other NSSS component evaluations.

- RCL piping loads
- Auxiliary line piping loads
- Primary equipment nozzle interface loads
- Primary equipment support foot pad loads
- Primary equipment support loads
- RCL piping and auxiliary line piping displacements
- Piping loads used in fracture mechanics evaluations
- Piping loads used in LBB evaluations
- Piping loads used in piping fatigue, piping stratification analyses, or time-limited aging analysis evaluations

It was also determined that there is no impact from the MUR-PU on the primary equipment supports (SG, RCP, and RV), because the loads on the supports have not changed. Seismic loads are unchanged since the MUR-PU does not affect the plant geometry or the input seismic excitations. Deadweight loads are unchanged since the hardware and its associated weight are not changing as part of the MUR-PU. The thermal and pipe break (i.e., LOCA) loads are not changing, as discussed in the paragraphs above, which states that the existing RCS piping analyses and resultant loads are not affected by the MUR-PU.

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

RESPONSE:

The MUR-PU operating conditions for the following BOP systems were reviewed for impact on the existing piping and supports design basis analyses:

- Main steam
- Feedwater
- Condensate
- FWH and moisture separator reheater (MSR) vents and drains
- Extraction steam
- Circulating water
- Component cooling water
- Auxiliary feedwater
- SFP cooling
- Service water
- SG blowdown
- Radwaste systems
- Reactor coolant piping
- Safety injection
- Chemical and volume control
- Residual heat removal
- Containment spray

Current and MUR-PU operating data (operating temperature, pressure, and flow rate) were obtained from heat balance diagrams, calculations, and/or other reference documents. Thermal, pressure, and flow rate "change factors" were determined, as required, to compare and evaluate changes in MUR-PU operating conditions. The "change factors" were based on the following ratios:

- The thermal "change factor" was based on the ratio of the MUR-PU to the current temperature. That is, the thermal change factor is (TMUR-PU-70°F) / (Tcurrent-70°F).
- The pressure "change factor" was determined by the ratio of (PressureMUR-PU / Pressurecurrent).
- The flow rate "change factor" was determined by the ratio of (Flow RateMUR-PU / Flow Ratecurrent).

These thermal, pressure, and flow rate change factors were used in determining piping systems acceptability for MUR-PU conditions. When the change factors are less than or equal to 1.0 (the current condition envelops or equals the MUR-PU condition), the piping system was considered acceptable for MUR-PU conditions. When the change factors are greater than 1.0, an evaluation was performed to address the specific temperature, pressure, and/or flow rate increase to document piping system acceptability.

The BOP piping systems reviewed remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from the MUR-PU conditions.

Containment systems are discussed in Section VI.I.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

IV.1.A.vi.a Steam Generator Thermal-Hydraulic Evaluation

RESPONSE:

A review of the changes in the T/H parameters for the MUR-PU shown in Table IV-1 indicates no impact to the key parameters. Evaluations show that all SG secondary side operating characteristics remain in an acceptable range for the MUR-PU and the relevant acceptance criteria are satisfied.

The best estimate moisture carryover (MCO) was reviewed and found to be within the MCO design limit of 0.1 percent based on the limiting operating conditions provided in Table IV-1.

IV.1.A.vi.b SG Structural Integrity

RESPONSE:

An evaluation was performed to determine if the MUR-PU operation conditions and transients change the acceptability of the various SG critical components with regard to structural limits, including fatigue, due to transients. Both primary side and secondary side SG components were examined.

Primary Side Evaluation

Primary side components consist of the divider plate, tubesheet-to-shell junction, tubesheet, and tube-to-tubesheet weld. The pressures in the SG system remain the same as prior to the uprate, and as discussed in Section IV.1, the MUR-PU will have no impact on current primary side NSSS design transients. Therefore, the primary and secondary stresses and fatigue evaluations will be unaffected.

Secondary Side Evaluation

Secondary side components consist of the feedwater nozzle, secondary side manway, secondary side manway fasteners, and steam nozzle. The only key change in NSSS design parameters from Table IV-1 is an increase in feedwater temperature of 2.6°F, to 446.0°F for the

MUR-PU. This increase also has some effect on the secondary side NSSS design transients, as discussed on Section IV.1.

This change could have an effect on the SG feedwater nozzle and thermal liner. But upon evaluation an increase in feedwater temperature of 2.6°F would actually result in a slightly smaller delta-T across the wall of the thermal liner resulting in slightly lower stress gradients and, therefore, slightly lower fatigue values. With such a small change, the difference would be slight, but the direction of change in values would be in the favorable direction. Threrefore, the existing analysis remains conservative.

The other critical secondary side components are all some distance away from the feedwater line, so there will be no significant effect due to the small increase in feedwater temperature. There are no impacts to the ASME stresses or fatigue usage for all the SG components.

IV.1.A.vi.c SG Tube Wear and FIV Evaluation

RESPONSE:

The impact of the MUR-PU on the SG tubes due to FIV and tube wear was evaluated based on the current design basis analysis and the changes in the T/H characteristics of the secondary side of the SG resulting from the uprate. The effects on the fluidelastic stability ratio and amplitudes of tube vibration due to turbulence were addressed. In addition, the effect of the uprate on potential future tube wear has been considered.

Fluidelastic Instability

The Farley Model 54F RSGs have an advanced anti-vibration bar (AVB) design, so it is most likely that the tubes would not be in an unsupported condition, which represents the bounding analyzed condition. Additionally, eddy current inspections of the Farley RSGs and SGs of a similar model have found only minor indications of tube wear at the AVB sites. The fluidelastic stability ratio calculated for the expected design condition would be applicable for this evaluation and would remain sufficiently below the allowable 1.0 for the limiting MUR-PU condition.

Turbulence

Previously calculated values of turbulence-induced displacements were assessed with the MUR-PU. The peak displacement is still well below the minimum displacement necessary for tube-to-tube contact. Hence, it can be concluded that the MU-PU will not result in unacceptably large turbulence amplitudes of vibration for either Farley Units 1 or 2 RSGs. Also, tube bending stresses due to turbulence are approximately two orders of magnitude below ASME fatigue endurance limits. Some increase in this stress limit due to MUR-PU will result in a stress that is still significantly below the minimum ASME fatigue endurance limit of 24 ksi and is acceptable, resulting in no addition to the fatigue usage factor for the tubes due to FIV.

Vortex Shedding

Additional consideration has been given to the effects of vortex shedding and FIV induced tube stress. Based upon the results obtained for both the turbulence response and tube wear, it can be concluded that any changes to the vortex shedding induced displacements or tube stress levels would be small because the basic relationships associated with these mechanisms are very similar to the turbulence mechanisms with respect to velocity and density. Since the vortex shedding displacements are small to begin with, additional small changes to these values would

not affect the conclusions of the prior analysis. Also, analyses indicated negligible tube response to potential vortex shedding consistent with expectations for two-phase flow over large tube arrays. As a result, it can be concluded that significant vortex shedding tube displacements and any FIV induced tube stresses would also be small.

Tube Wear

The tube wear rate will increase by some amount with the MUR-PU. However, only a small number of tubes in the Farley Units 1 and 2 RSGs have demonstrated any wear based on eddy current testing. All of this wear is in the vicinity of the AVBs and tube support plates and is evaluated in the condition monitoring and operational assessment (CMOA) reports prepared following inspections. A review of the latest available CMOA reports for Units 1 and 2 shows that tube wear based on the 95th percentile wear growth rate is acceptable based on the latest available inspection data. Therefore, tube integrity will not be compromised between inspections as a result of the MUR-PU Program.

Fatigue

An evaluation of the potential for high cycle fatigue of unsupported U-bend tubes was also performed. One of the prerequisites for high cycle U-bend fatigue is the formation of a dented tube support condition at the upper plate. This support condition is a result of a buildup of corrosion products associated with drilled holes in carbon steel tube support plates. The broached stainless steel support plate is designed to inhibit the development of corrosion products. If corrosion products were to build up, this could cause a tight support condition which could lead to high cycle tubing fatigue. Since the tight support condition is avoided by the current support plate design, high cycle fatigue at unsupported (no AVB support) inner row tubes is not predicted to occur in the Farley Units 1 and 2 RSGs and is not a concern.

Fatigue usage associated with general FIV resulting from the most limiting uprated operating condition was also assessed. It was found that the level of stress is still well below the endurance limit of approximately 24 ksi. Thereefore, the fatigue usage factor associated with FIV induced loadings while in the MUR-PU operating condition is negligible, and fatigue degradation for FIV is not anticipated for Farley Units 1 and 2 Model 54F RSG tubes.

Summary

In summary, analyses indicate that operation at the projected MUR-PU conditions will not result in rapid rates of tube wear or high levels of tube vibration to the general population of the tubes. Current tube wear conditions will increase as a result of the uprate and are addressed through the use of the uprate wear factor in the Degradation Assessment and CMOA as applicable. Monitoring the wear through eddy current inspections during outages will provide the basis for the remediation of the effects of tubes already experiencing wear and the basis to stabilize and/or plug tubes that exhibit wear in excess of design criterion limits.

IV.1.A.vii Reactor coolant pumps

RESPONSE:

Operating parameters, NSSS design transients, and piping loads are the inputs which could have an impact on the RCP AOR with the MUR-PU. Changes to these inputs were evaluated to

determine the impact of the MUR-PU on the RCP structural analysis and on the performance of the RCP motors.

The NSSS operating parameters in Table IV-1 which affect the RCP analysis are the reactor coolant pressure and T_{cold}. There are no changes to the current reactor coolant pressure of 2250 psia. The maximum MUR RCS T_{cold} of 540.5°F is less than the temperature evaluated in the AOR.

NSSS primary side design transients were determined to be unaffected by the MUR-PU, as discussed in the introduction of Section IV.1.

As discussed in Section IV.1.A.iv, loads applied to the RCP casing nozzles and support feet have not changed due to MUR-PU conditions.

The MUR-PU does not impact any inputs to the starting ability of the motor under the required conditions of reverse flow and minimum voltage. Therefore, performance of the RCP motors is unaffected by the MUR-PU.

It can be concluded that the RCPs at Farley Units 1 and 2 are shown to be acceptable for operation at MUR-PU conditions.

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

RESPONSE:

The MUR-PU operating conditions were reviewed for impact on the existing pressurizer design basis analysis for Farley Units 1 and 2. The limiting operating conditions of the pressurizer occurs when the RCS pressure is high and the RCS T_{hot} and T_{cold} temperatures are low. This maximizes the Delta-T in the pressurizer. The limiting T_{hot} and T_{cold} conditions for MUR-PU did not change from the current operation.

Due to the flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS at T_{hot}. By qualifying the surge and spray nozzle all other subcomponents of the pressurizer are also qualified. As such, these evaluations were performed to support the MUR-PU operation to address the impact on the pressurizer.

The evaluation showed the changes in T_{hot} and T_{cold} are enveloped by the AOR parameters.

IV.1.A.ix Safety related valves

RESPONSE:

The revised operating conditions were reviewed for impact on the design basis of existing safety-related valves. No changes in RCS design or operating pressure were made as part of the power uprate. The evaluations concluded that the temperature changes due to the power uprate have, at most, an insignificant effect on the differential pressures used in the existing analyses. Safety-related valves were reviewed within the applicable system (Section VI) and program (Section VII.6.D) evaluations. None of the safety-related valves in the NSSS required a change to their design or operation as a result of the MUR power uprate.

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

References for Section IV.1.A

IV.1.A.1 Westinghouse Report, WCAP-9620, Revision 1, "Reactor Internals Heat Generation Rates and Neutron Fluences," December 1983.

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

IV.1.B.i <u>Stresses</u>

RESPONSE:

Evaluations were performed to demonstrate that the revised design conditions and design transients for the NSSS components, piping, and interface systems were within the existing structural design basis analyses. Stress evaluations and any impacts are discussed within the subsections under Section IV.1.A.

IV.1.B.ii <u>Cumulative usage factors</u>

RESPONSE:

Evaluations were performed to demonstrate that the revised design conditions and design transients for the NSSS components, piping, and interface systems were within the existing structural design basis analyses. Any impacts to cumulative usage factors or fatigue evaluations are discussed within the subsections under Section IV.1.A.

IV.1.B.iii <u>Flow induced vibration</u>

RESPONSE:

The impact of FIV on the RV internals components were evaluated in Section IV.1.A.ii.b and found to be acceptable. The impact of FIV on the SG components is discussed in Section IV.1.A.vi.c and was also found to be acceptable.

IV.1.B.iv <u>Changes in temperature (pre- and post-uprate)</u>

RESPONSE:

See the discussion in Section IV.1 regarding the small change in operating temperature limits (T_{hot} and T_{cold}) and how this was considered in the various component analyses.

There is no change to the RCS average temperature limit in TS 3.4.1 and the COLR.

IV.1.B.v <u>Changes in pressure (pre- and post-uprate)</u>

RESPONSE:

There will be no change in RCS operating pressure as a result of the MUR power uprate. The nominal operating pressure is 2250 psia (Table IV-1). There is no change to the RCS pressure limit in TSs 2.1.2 or 3.4.1.

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

RESPONSE:

The MUR-PU has no effect on the operating RCS flow rates. The TDF and MDF shown in Table IV-1 remain unchanged for the power uprate.

IV.1.B.vii <u>High and moderate energy line break (HELB) locations</u>

RESPONSE:

The affected piping systems as listed in Section IV.1.A.v were evaluated to address revised MUR-PU operating conditions. Applicable pipe rupture postulation criteria were reviewed; and changes to piping operating temperatures, pressures, and piping system stress levels resulting from the MUR-PU were reviewed against pipe break evaluation requirements. Because there was no adverse impact as a result of the MUR-PU to high/moderate energy piping systems in areas with safety related components, there will also be no impact to applicable pipe break evaluations.

In summary, the evaluations performed for applicable piping systems did not result in any new or revised break/crack locations; and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for MUR-PU conditions.

IV.1.B.vii.a Leak Before Break Evaluation

RESPONSE:

The LBB analyses of the primary RCL and pressurizer surge line piping were evaluated for the changes associated with the Farley Units 1 and 2 MUR-PU. There were no physical plant modifications to the RCL and surge line piping systems that impacted the piping stress analyses. The changes were limited to operational changes, such as those associated with the temperature, pressure, and flow characteristics of the reactor coolant. The evaluation of these LBB analyses for the MUR-PU considered the inputs used in the analyses that are potentially impacted by operational changes, specifically the NSSS design parameters, NSSS design transients, and RCS primary loop piping loads.

The evaluation concluded that the only change from these inputs was a minor change to the RCS temperatures, which has negligible impact on the LBB analysis. The NSSS design transients and RCS piping loads are not affected by the MUR-PU. Therefore, the existing RCL and surge line piping LBB analyses for Farley Units 1 and 2 remain applicable and valid for the MUR-PU.

IV.1.B.vii.b Jet impingement and thrust forces

RESPONSE:

The affected piping systems as listed in Section IV.1.A.v were evaluated to address revised MUR-PU operating conditions for line breaks resulting in jet impingements and pipe whip. Applicable pipe rupture postulation criteria were reviewed and changes to piping operating temperatures, pressures, and piping system stress levels resulting from the MUR-PU were reviewed against pipe break evaluation requirements. Since there was no adverse impact as a result of the MUR-PU to high/moderate energy piping systems in areas with safety related components, there will also be no impact to applicable pipe break evaluations.

In summary, the evaluations performed for applicable piping systems did not result in any new or revised break/crack locations and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for MUR-PU conditions.

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

<u>Reactor vessel integrity</u>

RESPONSE:

RV integrity is impacted by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence due to the MUR-PU have been evaluated to determine the impact on the Farley Units 1 and 2, 54 effective full-power years (EFPYs) heatup and cooldown limit curves presented in References IV.1.C.1 and IV.1.C.2, respectively. In addition, the review evaluated the impacts of the MUR-PU on the AOR for the reference temperature for pressurized thermal shock (RT_{PTS}) values (10 CFR 50.61), the upper-shelf energy (USE) values, and the material surveillance capsule withdrawal schedule. Each of these areas is described in subsequent sections.

The most critical area, in terms of RV integrity, is the beltline region of the RV. The beltline region is defined in the ASTM (Reference IV.1.C.3) as "the irradiated region of the RV (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material." This definition is consistent with the definition of beltline in Reference IV.1.C.4.

The Farley Units 1 and 2 beltline materials traditionally included intermediate shell plates, lower shell plates, the intermediate to lower shell circumferential weld, and intermediate/lower shell longitudinal welds at each Unit; however, as described in NRC RIS 2014-11 (Reference IV.1.C.5), any RV materials that are predicted to experience a neutron fluence exposure greater than 1.0×10^{17} n/cm^2 (E > 1.0 MeV) at the end of the licensed operating period should be considered in the development of pressure-temperature (P-T) limit curves. The additional materials that exceed this fluence threshold are referred to as extended beltline materials and are evaluated to guarantee that the applicable acceptance criteria are met. The extended beltline materials include upper shell forging, upper to intermediate shell circumferential weld, the inlet nozzle forgings, and the inlet nozzle to upper shell welds. Note, the outlet nozzle

forgings and the outlet nozzle to upper shell welds are projected to experience a fluence of less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV), but are treated as extended beltline material herein, consistent with the Farley Units 1 and 2 AOR. In addition to the effects of irradiation, Reference IV.1.C.5 requires that the effect of higher stress intensities in the nozzle materials be evaluated for the effects on the P-T limit curves, regardless of exposure.

Refer to Attachment 3 for justification of application of Reference IV.1.C.7 in the extended beltline region.

IV.1.C.i Pressurized thermal shock calculations

RESPONSE:

A limiting condition on RV integrity, known as pressurized thermal shock (PTS), may occur during a severe system transient such as a LOCA or a steam line break. These transients may challenge the integrity of a RV under the following conditions:

- Severe overcooling of the inside surface of the vessel wall followed by high repressurization
- Significant degradation of vessel material toughness caused by radiation embrittlement
- The presence of a critical-size defect in the vessel wall

PTS could be an issue if one of these transients acts on the beltline region of a RV where a reduced fracture resistance exists due to neutron irradiation. This type of event may result in the propagation of flaws that are postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The PTS Rule (10 CFR 50.61, Reference IV.1.C.6) established screening criteria on PWR vessel embrittlement as measured by the RT_{PTS}. The RT_{PTS} screening criteria values were established (using conservative fracture mechanics analysis techniques) for axial welds, plates, and circumferential weld seams for end-of-license plant operation.

The PTS calculations were performed using the latest procedures specified by the NRC in the PTS Rule (Reference IV.1.C.6) for 54 EFPY. To evaluate the effects of the MUR-PU, the PTS values for the Farley Units 1 and 2 beltline and extended beltline materials were calculated using the MUR-PU fluences (see Section IV.1.C.ii). A summary of the limiting beltline and extended beltline material RT_{PTS} values is presented in Tables IV.1.C-1 and IV.1.C-2. All RT_{PTS} values remain below the NRC screening criteria values using the projected uprated fluence values at 54 EFPY.

Table IV.1.C-1: RT_{PTS} Calculations for Farley Unit 1 Beltline and Extended Beltline Region Materials at 54 EFPY

Material	CF (°F)	Fluence (n/cm^2, E > 1.0 MeV)	FF[1]	$RT_{NDT(U)}$ (°F)	ΔRT_{PTS}[2] (°F)	Margin[3] (°F)	RT_{PTS}[4] (°F)
Inlet Nozzle B6917-1	141.45	0.0157	0.148	-18	20.9	20.9	23.8

Material	CF (°F)	Fluence (n/cm^2, E > 1.0 MeV)	FF(1)	$RT_{NDT(U)}$ (°F)	ΔRT_{PTS}(2) (°F)	Margin(3) (°F)	RT_{PTS}(4) (°F)
Inlet Nozzle B6917-2	141.0	0.0157	0.148	29	20.8	20.8	70.7
Inlet Nozzle B6917-3	142.05	0.0157	0.148	-48	21.0	21.0	-6.0
Outlet Nozzle B6916-1	139.95	0.00948	0.106	-17	14.8	14.8	12.6
Outlet Nozzle B6916-2	140.3	0.00948	0.106	-29	14.8	14.8	0.7
Outlet Nozzle B6916-3	140.3	0.00948	0.106	-23	14.8	14.8	6.7
Upper Shell Forging B6914	120.1	0.820	0.944	30	113.4	34.0	177.4
Intermediate Shell Plate B6903-2	91.0	6.05	1.438	0	130.9	34.0	164.9
Intermediate Shell Plate B6903-3	82.2	6.05	1.438	10	118.2	34.0	162.2
Lower Shell Plate B6919-1	97.8	6.03	1.437	15	140.6	34.0	189.6
Lower Shell Plate B6919-1 with Non-Credible Surveillance Data	106.46	6.03	1.437	15	153.0	34.0	**202.0**
Lower Shell Plate B6919-2	98.2	6.03	1.437	5	141.2	34.0	180.2
Inlet/Outlet Nozzle to Upper Shell Girth Weld Seam (1-897 A → F) (Multiple Heats)	54.0	0.0157	0.148	10	8.0	34.9	52.9
Upper to Intermediate Shell Circumferential Weld Seam 10-894 (Heat # 90099)	91.4	0.925	0.978	-56	89.4	65.5	98.9
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	126.3	1.83	1.166	-56	147.2	65.5	156.7
Intermediate Shell Longitudinal Weld Seams 19-894 A & B with Credible Surveillance Data (Heat # 33A277)	120.52	1.83	1.166	-56	140.5	44.0	128.5
Intermediate to Lower Shell Circumferential Weld Seam 11-894 (Heat # 6329637)	98.4	6.02	1.437	-56	141.4	65.5	150.9
Lower Shell Longitudinal Weld Seams 20-894 A & B (Heat # 90099)	91.4	1.85	1.169	-56	106.8	65.5	116.3

Material	CF (°F)	Fluence (n/cm^2, E > 1.0 MeV)	FF(1)	$RT_{NDT(U)}$ (°F)	ΔRT_{PTS}(2) (°F)	Margin(3) (°F)	RT_{PTS}(4) (°F)

Notes:

1. FF = fluence factor = $f^{(0.28 - 0.1 \log(f))}$
2. ΔRT_{PTS} = CF * FF.
3. Per Reference IV.1.C.1, margin = $2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$; where:
   - σ_U = 0°F when $RT_{NDT(U)}$ values are based on measured data, or
     σ_U = 17°F when generic mean values from Reference IV.1.C.1 are used.
   - σ_Δ = 17°F for the base metal when capsule data are not credible or not used, or
     σ_Δ = 8.5°F for base metal when credible surveillance capsule data are used.

     However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
   - σ_Δ = 28°F for the weld metal when capsule data are not credible or not used, or
     σ_Δ = 14°F for weld metal when credible surveillance data is used.

     However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
4. RT_{PTS} = $RT_{NDT(U)}$ + ΔRT_{PTS} + Margin

Table IV.1.C-2: RT_{PTS} Calculations for Farley Unit 2 Beltline and Extended Beltline Region Materials at 54 EFPY

Material	CF (°F)	Fluence (n/cm², E > 1.0 MeV)	FF[1]	$RT_{NDT(U)}$ (°F)	ΔRT_{PTS}[2] (°F)	Margin[3] (°F)	RT_{PTS}[4] (°F)
Inlet Nozzle B7218-1	137.85	0.0164	0.152	-55	20.9	20.9	-13.1
Inlet Nozzle B7218-2	136.8	0.0164	0.152	-55	20.8	20.8	-13.4
Inlet Nozzle B7218-3	138.2	0.0164	0.152	-60	21.0	21.0	-18.0
Outlet Nozzle B7217-1	138.55	0.00988	0.109	-47	15.1	15.1	-16.9
Outlet Nozzle B7217-2	138.2	0.00988	0.109	-71	15.0	15.0	-40.9
Outlet Nozzle B7217-3	138.2	0.00988	0.109	-43	15.0	15.0	-12.9
Upper Shell Forging B7216-1	121.1	0.850	0.954	30	115.6	34.0	179.6
Intermediate Shell Plate B7203-1	100.0	6.00	1.437	15	143.7	34.0	192.7
Intermediate Shell Plate B7212-1	149.0	6.00	1.437	-10	214.1	34.0	238.1
Intermediate Shell Plate B7212-1 with Credible Surveillance Data	144.37	6.00	1.437	-10	207.4	17.0	**214.4**
Lower Shell Plate B7210-1	89.8	6.00	1.437	18	129.0	34.0	181.0
Lower Shell Plate B7210-2	98.7	6.00	1.437	10	141.8	34.0	185.8
Inlet/Outlet Nozzle to Upper Shell Girth Weld Seams (1-926 A → F) (Multiple Heats)	95.0	0.0164	0.152	10	14.4	36.9	61.4
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 5P5622)	74.1	0.958	0.988	-40	73.2	56.0	89.2
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 51922)	68.0	0.958	0.988	-56	67.2	65.5	76.7
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 3P4767)	46.3	0.958	0.988	-56	45.7	57.0	46.7
Intermediate Shell Longitudinal Weld Seam 19-923 A (Heat # HODA)	36.8	1.85	1.169	-56	43.0	54.8	41.8

Material	CF (°F)	Fluence (n/cm², E > 1.0 MeV)	FF[1]	$RT_{NDT(U)}$ (°F)	ΔRT_{PTS}[2] (°F)	Margin[3] (°F)	RT_{PTS}[4] (°F)
Intermediate Shell Longitudinal Weld Seams 19-923 A & B (Heat # BOLA)	36.8	1.85	1.169	-60	43.0	43.0	26.0
Intermediate Shell Longitudinal Weld Seams 19-923 A &B with Non-Credible Surveillance Data (Heat # BOLA)	20.61	1.85	1.169	-60	24.1	24.1	-11.8
Intermediate to Lower Shell Circumferential Weld Seam 11-923 (Heat # 5P5622)	74.1	6.00	1.437	-40	106.5	56.0	122.5
Lower Shell Longitudinal Weld Seams 20-923 A & B (Heat # 83640)	37.3	1.87	1.171	-70	43.7	43.7	17.4

Notes:

1. FF = fluence factor = $f^{(0.28 - 0.1 \log(f))}$
2. ΔRT_{PTS} = CF * FF.
3. Per 10 CFR 50.61, margin = $2\sqrt{\sigma_U^2 + \sigma_\Delta^2}$; where:
   - σ_U = 0°F when $RT_{NDT(U)}$ values are based on measured data, or
     σ_U = 17°F when generic mean values from Reference 14 are used.
   - σ_Δ = 17°F for the base metal when capsule data are not credible or not used, or
     σ_Δ = 8.5°F for base metal when credible surveillance capsule data are used.

     However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
   - σ_Δ = 28°F for the weld metal when capsule data are not credible or not used, or
     σ_Δ = 14°F for weld metal when credible surveillance data is used.

     However, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
4. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin

IV.1.C.ii Fluence evaluation

RESPONSE:

In the assessment of the state of embrittlement of light water reactor (LWR) pressure vessels, an accurate evaluation of the neutron exposure of each of the materials comprising the beltline region of the vessel is required. In Appendix G to 10 CFR 50, the beltline region is defined as:

> *The region of the reactor vessel shell material (including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the reactor core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The beltline region is comprised of two ring forgings (forming the lower shell course and the intermediate shell course). The two ring pieces are joined by a circumferential weld. Each of these materials must be considered in the overall embrittlement assessments of the beltline region. Therefore, plant-specific exposure assessments must include evaluations as a function of axial and azimuthal location over the entire beltline region.

A neutron fluence assessment was performed for the Farley Units 1 and 2 pressure vessel beltline region for the MUR-PU following the NRC-accepted methodology from Reference IV.1.C.7, which complies with guidance established in Reference IV.1.C.8. In this assessment, fast neutron exposures expressed in terms of fast neutron fluence ($E > 1.0$ MeV) and iron atom displacements (dpa) were established for each of the materials comprising the beltline region of the pressure vessel.

Following completion of these plant-specific exposure assessments encompassing previously completed fuel cycles for each unit, projections of the future neutron exposure of the pressure vessel beltline materials extending to 60 EFPY of operation were performed. In performing the fluence projections, the assumption was made that the spatial power distributions and reactor operating characteristics anticipated for the MUR-PU would be representative through the 60 EFPY operating period.

In performing the fast neutron exposure evaluations, plant-specific forward transport calculations were carried out using a 3-D discrete ordinates transport method (Reference IV.1.C.7) to directly solve for the space- and energy-dependent scalar fluence rate, $\phi(r,\theta,z,E)$.

All of the transport calculations were carried out using the RAPTOR-M3G Version 2.0 3-D discrete ordinates code and the BUGLE-96 cross-section library (Reference IV.1.C.9). The BUGLE-96 library provides a coupled 47-neutron and 20-gamma-ray group cross-section data set produced specifically for LWR applications. In these analyses, anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{12} order of angular quadrature. The degree of Legrendre expansion and angular discretization, along with mesh sizing, were determined to be appropriate such that further parameter refinement resulted in a less than 2 percent change in beltline region results, consistent with the application of the methodology in Reference IV.1.C.7.

Three-dimensional r,θ,z geometrical models were built in an octant core geometry to include all the possible configurations of neutron pads. In addition to the core, reactor internals, RPV, and primary biological shield, the models developed for this geometry also included explicit representations of the stainless steel former plates, surveillance capsules, the RPV cladding, and the reflective insulation.

The analytical model extended radially from the centerline of the reactor core out to a location interior to the primary biological shield and over an axial span from an elevation approximately 5.5 feet below the active fuel to approximately 4.5 feet above the active fuel.

In developing the r,θ,z analytical models of the reactor geometry, nominal design dimensions were employed for the various structural components. Cycle-dependent water temperatures and, hence, coolant densities in the reactor core and downcomer regions of the reactor were taken to be representative of full-power operating conditions. The reactor core was treated as a homogeneous mixture of fuel, cladding, water, and miscellaneous core structures such as fuel

assembly grids, guide tubes, etc. Mesh sizes for the r,θ,z geometric models were chosen to assure that proper convergence of the inner iterations was achieved on a pointwise basis. The pointwise inner iteration fluence rate convergence criterion used in the calculations was 0.001, consistent with guidance in Reference IV.1.C.8.

The plant and fuel-cycle-specific calculated fast neutron (E > 1.0 MeV) fluence values and integrated iron dpa experienced by the materials comprising the beltline regions of the RPV were calculated. This data represents the maximum neutron exposures at the pressure vessel clad/base metal interface at azimuthal angles of 0°, 15°, 30°, and 45° relative to the core major axes, for fuel cycles completed to date. In addition, the neutron exposure projections were based on the spatial core power distributions and associated plant operating characteristics of Unit 2 Cycle 25, adjusted to account for the MUR-PU, including an analyzed core thermal power of 2831 MWt.

In addition, calculated fast neutron (E > 1.0 MeV) fluence and iron atom displacement values for the pressure vessel shell, circumferential welds, nozzle welds, and intermediate shell and lower shell plate longitudinal welds were determined.

The results were utilized as input to the RV integrity analyses described in Sections IV.1.C.i, IV.1.C.iii, IV.1.C.v, and IV.1.C.vi.

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

RESPONSE:

The Farley Units 1 and 2 Pressure-Temperature Limits Report (PTLR) (Reference IV.1.C.10) currently implements 54 EFPY P-T limit curves which were developed in References IV.1.C.1 and IV.1.C.2, respectively. The development of the P-T limit curves is consistent with the methodologies of References IV.1.C.11 and IV.1.C.12 and complies with Reference IV.1.C.4. The P-T limit curves take into account the limiting $RT_{NDT(U)}$ value of the closure head flange and vessel flange regions as required by Reference IV.1.C.4. The cooldown curves were developed for cooldown rates of 0, 20, 40, 60, and 100°F/hr. The heatup curves were developed for heatup rates of 60 and 100°F/hr.

References IV.1.C.13 and IV.1.C.14 contain an evaluation of the nozzle materials effect on P-T limit curves for both Units 1 and 2, consistent with Reference IV.1.C.5, and determined the nozzle materials are non-bounding. The analysis conservatively included the embrittlement effects on the nozzles even if nozzles projected fluence values were less than 1.0 x 10^{17} n/cm^2 (E > 1.0 MeV) at 54 EFPY. The effects of the MUR-PU on these embrittlement effects were evaluated to guarantee that the evaluation of the nozzle P-T limit remains valid. Consistent with the AOR, the fluence value for Farley Unit 1 inlet nozzle B6917-2 was conservatively taken at a higher elevation than the other nozzles. This location is consistent with AOR and conservatively below the postulated flaw crack tip. Use of the fluence value at a more realistic elevation of the postulated crack tip was accepted by the NRC for Farley Unit 1 inlet nozzle B6917-2 (Reference IV.1.C.15). Therefore, using a fluence at 11 cm above the nozzle to upper shell weld is justified.

The heatup and cooldown P-T limit curves were generated in accordance with References IV.1.C.1 and IV.1.C.2 using the most limiting adjusted reference temperature (ART) values for the RV materials calculated at that time. The 1/4T and 3/4T ART values were recalculated per

the methodology in Reference IV.1.C.11, taking into account the new 3-D fluence model with the MUR-PU operating conditions. The limiting ART values based on the MUR-PU conditions are summarized in Table IV.C.1-1.

Table IV.1.C-3 shows that the limiting ART values for Farley Units 1 and 2 at the 1/4T and 3/4T locations under MUR-PU conditions will exceed the ART values used in the current P-T limit curves at end-of-license extension (EOLE). For Unit 1, ART values used in the P-T limit curves will be reached when the fluence of lower shell plate B6919-1 reaches 5.80E+19 n/cm^2 (E > 1.0 MeV). This will occur at 51.9 EFPY. For Unit 2, ART values used in the P-T limit curves will be reached when the surface fluence of the intermediate shell plate B7212-1 reaches 5.80E+19 n/cm^2 (E > 1.0 MeV). This will occur at 52.1 EFPY.

References IV.1.C.1 and IV.1.C.2 do not consider the effects of the higher stresses associated with the geometry of the nozzles required by Reference IV.1.C.5. The effects of the nozzles higher stress intensities on the P-T limit curves were considered in References IV.1.C.13 and IV.1.C.14. In order to guarantee the conclusions of these documents remain valid, Table IV.1.C-4 compares the updated ART values of the nozzles given the MUR-PU to those used in the nozzle P T limit curve evaluation.

Table IV.1.C-4 shows that the limiting nozzle ART values for Units 1 and 2 will remain bounded at EOLE under MUR-PU conditions by the ART values used in the AOR for nozzle P-T limit curves. Therefore, the previous conclusion that the nozzle P-T limit curves will not be limiting compared to the EOLE P-T limit curves remains valid.

In conclusion, as a result of the new 3-D fluence model input and the MUR-PU operating conditions, the term of applicability of the P-T limit curves currently implemented in the PTLR will need to be reduced due to the increase in the limiting beltline ART values at 54 EFPY. The current Unit 1 P-T limit curves will be valid through 51.9 EFPY. The current Unit 2 P-T limit curves will be valid through 52.1 EFPY.

Table IV.1.C-3: Summary of the Limiting ART Values Used in the Generation of the Farley Units 1 and 2 Heatup and Cooldown Curves at 54 EFPY

Unit	1/4T Limiting ART		3/4T Limiting ART	
	AOR(1)	Under MUR-PU Conditions	AOR(1)	Under MUR-PU Conditions
1	191°F	191.9°F	166°F	166.6°F
	Lower Shell Plate B6919-1 (Position 2.1 with non-credible surveillance data)			
2	200°F	200.7°F	165°F	166.3°F
	Intermediate Shell Plate B7212-1 (Position 2.1 with credible surveillance data)			

Notes:

1. The limiting ART values used to develop the P-T limit curves in the AOR are from References IV.1.C.1 and IV.1.C.2 for Units 1 and 2, respectively, and are consistent with the PTLRs.

Table IV.1.C-4: Summary of the Limiting ART Values Used in the Generation of the Farley Units 1 and 2 Nozzle Heatup and Cooldown Curves at 54 EFPY

Unit	Component	AOR ART[1] (°F)	Revised ART Under MUR-PU Conditions (°F)
1	Inlet Nozzle	55.1	51.1
	Outlet Nozzle	29.8	12.6
2	Inlet Nozzle	16.6	-13.1
	Outlet Nozzle	18.3	-12.9

Notes:

1. The limiting ART values used to develop the nozzle P-T limit curves in the AOR are from References IV.1.C.13 and IV.1.C.14 for both Units 1 and 2.

IV.1.C.iv **Low-temperature overpressure protection**

RESPONSE:

The low-temperature overpressure protection system (LTOPS), also known as the cold overpressure mitigation system (COMS), provides RCS pressure relief capability during relatively low temperature operation (i.e., RCS temperature less than about 350°F). The LTOPS provides capability to mitigate the overpressure transients that may occur during cold shutdown, heatup, and cooldown operations to minimize the potential for challenging RV integrity when operating at or near RV ductility limits (i.e., 10 CFR 50, Appendix G limits).

The potential overpressurization transients are categorized as either mass injection (MI) or heat injection (HI). The limiting design basis HI transient is defined as the start of an idle RCP with a maximum temperature asymmetry between the RCS and SGs whereby the SG secondary side is a maximum of 50°F hotter than the RCS primary side. The limiting design basis MI transient is defined as having a limiting coolant input capability of a maximum of one charging pump capable of injection into the RCS when one or more RCS cold leg temperatures are ≤ 180°F or a maximum of two charging pumps capable of injection into the RCS when all the RCS cold leg temperatures are >180°F and the accumulators are isolated.

The design basis LTOPS HI transient is initiated in Modes 4, 5, or 6 and therefore is not affected by changes to the core power level and corresponding changes in the NSSS design parameters shown in Table IV-1. The MUR-PU also does not result in changes to key HI transient input parameters such as RCP startup characteristics, RCS fluid volumes, SG heat transfer areas, or SG secondary side water mass at shutdown conditions. Therefore, the required LTOPS relief capacity during the HI transient remains applicable at MUR-PU conditions.

Additionally, the P-T limits evaluations documented in section IV.1.C.iii showed that as a result of a new fluence model and the MUR-PU operating conditions, the term of applicability of the P-T limit curves currently implemented in the PTLR needs to be reduced from 54 EFPY to 51.9 EFPY for Unit 1 and 52.1 EFPY for Unit 2. Similarly, the current calculated LTOPS enable temperatures remain valid, but the applicability for the enable temperatures are reduced consistent with the P-T limit.

IV.1.C.v **Upper shelf energy**

RESPONSE:

The integrity of the RV may be affected by changes in system temperatures and pressures resulting from the power uprate. To address this consideration, an evaluation is performed to assess the impact of the MUR-PU on the USE values for all the RV beltline and extended beltline materials in the Farley Units 1 and 2 RV at EOLE. The USE assessment uses the results of the neutron fluence evaluation for MUR-PU and Figure 2 of Reference IV.1.C.11 to determine if a further decrease in USE at EOLE would occur due to the effects of the MUR-PU on the fluence projections.

Based on the current analysis, all beltline and extended beltline materials are expected to have a USE greater than 50 ft-lb through EOLE (54 EFPY) as required by Reference IV.1.C.4. To evaluate the effects of the MUR-PU, the EOLE (54 EFPY) USE values were predicted using the MUR-PU 1/4T fluence projections at EOLE along with Figure 2 of Reference IV.1.C.11. The MUR-PU USE values are summarized for the beltline and extended beltline materials in Tables IV.1.C-5 and IV.1.C-6. Under MUR-PU conditions, the EOLE USE values for all reactor beltline and extended beltline materials meet the requirements of Reference 4, that the USE is greater than 50 ft-lb in each case.

Table IV.1.C-5: USE Projections for Farley Unit 1 Beltline and Extended Beltline Materials at 54 EFPY

Material	1/4T Fluence(1) (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE (ft-lb)	Projected USE Decrease(2) (%)	Projected USE (ft-lb)
Position 1.2				
Inlet Nozzle B6917-1	0.0157(3)	110	11	97.9
Inlet Nozzle B6917-2	0.0157(3)	80	11	71.2
Inlet Nozzle B6917-3	0.0157(3)	98	11	87.2
Outlet Nozzle B6916-1	0.00948(3)	96.5	11	85.9
Outlet Nozzle B6916-2	0.00948(3)	97.5	11	86.8
Outlet Nozzle B6916-3	0.00948(3)	100	11	89.0
Upper Shell Forging B6914	0.511	95.3	22	74.3
Intermediate Shell Plate B6903-2	3.77	99	30	69.3
Intermediate Shell Plate B6903-3	3.77	87	29	61.8
Lower Shell Plate B6919-1	3.76	86	32	58.5
Lower Shell Plate B6919-2	3.76	86	32	58.5
Inlet/Outlet Nozzle to Upper Shell Girth Weld Seams (1-897 A → F) (Multiple Heats)	0.0157(3)	73	8	67.2
Upper to Intermediate Shell Circumferential Weld Seam 10-894 (Heat # 90099)	0.577	82.5	31	56.9
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	1.14	149	42	86.4
Intermediate to Lower Shell Circumferential Weld Seam 11-894 (Heat # 6329637)	3.75	104	47	55.1
Lower Shell Longitudinal Weld Seams 20-894 A & B (Heat # 90099)	1.15	82.5	36	52.8

Material	1/4T Fluence(1) (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE (ft-lb)	Projected USE Decrease(2) (%)	Projected USE (ft-lb)
Position 2.2(4)				
Lower Shell Plate B6919-1	3.76	86	26	63.6
Intermediate Shell Longitudinal Weld Seams 19-894 A & B (Heat # 33A277)	1.14	149	20	119.2

Notes:

1. The 1/4T fluence was calculated using the surface fluence, the equation $f = f_{surf} * e^{-0.24\,(x)}$ from Reference IV.1.C.11, and the Farley Units 1 and 2 reactor vessel beltline wall thickness of 7.875 inches.
2. Percentage USE decrease values are based on Reference IV.1.C.11. The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Reference IV.1.C.11 and using the material-specific Cu wt. percent values. Base metal and weld Cu wt. percent lines were extended into the low fluence area of Figure 2 of Reference IV.1.C.11, i.e., below 10^{18} n/cm^2, in order to determine the USE percent decrease as needed. For Cu wt. percent values below the minimum line on Figure 2 of Reference IV.1.C.11, the minimum line was conservatively used.
3. The fluence for the inlet/outlet nozzles and nozzle to upper shell welds is conservatively taken at the surface, neglecting attenuation through the vessel wall. Fluence values below 2 x 10^{17} n/cm^2 (E > 1.0 MeV) were rounded to 2 x 10^{17} n/cm^2 (E > 1.0 MeV) when determining the percent decrease because 2 x 10^{17} n/cm^2 is the lowest fluence displayed in Figure 2 of Reference IV.1.C.11.
4. Calculated using surveillance capsule measured percent decrease in USE and Reference IV.1.C.11, Position 2.2.

Table IV.1.C-6: USE Projections for Farley Unit 2 Beltline and Extended Beltline Materials at 54 EFPY

Material	1/4T Fluence[(1)] (x 10^{19} n/cm^2, E > 1.0 MeV)	Unirradiated USE (ft-lb)	Projected USE Decrease[(2)] (%)	Projected USE (ft-lb)
Position 1.2				
Inlet Nozzle B7218-1	0.0164[(3)]	112	11	99.7
Inlet Nozzle B7218-2	0.0164[(3)]	103	11	91.7
Inlet Nozzle B7218-3	0.0164[(3)]	98	11	87.2
Outlet Nozzle B7217-1	0.00988[(3)]	100	11	89.0
Outlet Nozzle B7217-2	0.00988[(3)]	108	11	96.1
Outlet Nozzle B7217-3	0.00988[(3)]	103	11	91.7
Upper Shell Forging B7216-1	0.530	96.2	22	75.0
Intermediate Shell Plate B7203-1	3.74	100	32	68.0
Intermediate Shell Plate B7212-1	3.74	100	40	60.0
Lower Shell Plate B7210-1	3.74	103	30	72.1
Lower Shell Plate B7210-2	3.74	99	32	67.3
Inlet/Outlet Nozzle to Upper Shell Girth Weld Seams (1-926 A → F) (Multiple Heats)	0.0164[(3)]	97	9	88.3
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 5P5622)	0.597	102	27	74.5
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 51922)	0.597	101	17	83.8
Upper to Intermediate Shell Circumferential Weld Seam 10-923 (Heat # 3P4767)	0.597	101	21	79.8
Intermediate Shell Longitudinal Weld Seam 19-923 A (Heat # HODA)	1.15	131	20	104.8
Intermediate Shell Longitudinal Weld Seams 19-923 A & B (Heat # BOLA)	1.15	148	20	118.4

Material	1/4T Fluence(1) (x 10^{19} n/cm², E > 1.0 MeV)	Unirradiated USE (ft-lb)	Projected USE Decrease(2) (%)	Projected USE (ft-lb)
Intermediate to Lower Shell Circumferential Weld Seam 11-923 (Heat # 5P5622)	3.74	102	40	61.2
Lower Shell Longitudinal Weld Seams 20-923 A & B (Heat # 83640)	1.17	126	20	100.8
Position 2.2(4)				
Intermediate Shell Plate B7212-1	3.74	100	42	58.0
Intermediate Shell Longitudinal Weld Seams 19-923 A & B (Heat # BOLA)	1.15	148	10	133.2

Notes:

1. The 1/4T fluence was calculated using the surface fluence, the equation $f = f_{surf} * e^{-0.24(x)}$ from Reference IV.1.C.11, and the Farley Units 1 and 2 reactor vessel beltline wall thickness of 7.875 inches.
2. Percentage USE decrease values are based on Reference IV.1.C.11. The Position 1.2 USE decrease values were calculated by plotting the 1/4T fluence values on Figure 2 of Reference IV.1.C.11 and using the material-specific Cu wt. percent values. Base metal and weld Cu wt. percent lines were extended into the low fluence area of Figure 2 of Reference IV.1.C.11, i.e., below 10^{18} n/cm², in order to determine the USE percent decrease as needed. For Cu wt. percent values below the minimum line on Figure 2 of Reference IV.1.C.11, the minimum line was conservatively used.
3. The fluence for the inlet/outlet nozzles and nozzle to upper shell welds is conservatively taken at the surface, neglecting attenuation through the vessel wall. Fluence values below 2×10^{17} n/cm² (E > 1.0 MeV) were rounded to 2×10^{17} n/cm² (E > 1.0 MeV) when determining the percent decrease because 2×10^{17} n/cm² is the lowest fluence displayed in Figure 2 of Reference IV.1.C.11.
4. Calculated using surveillance capsule measured percent decrease in USE and Reference IV.1.C.11, Position 2.2.

IV.1.C.vi Surveillance capsule withdrawal schedule

RESPONSE:

A surveillance capsule withdrawal schedule is developed to periodically remove surveillance capsules from the RV to effectively monitor the condition of the RV materials under actual operating conditions. Reference IV.1.C.3 defines the recommended number of surveillance capsules and the recommended withdrawal schedule, based on the vessel material predicted transition temperature shifts (ΔRT_{NDT}). The surveillance capsule withdrawal schedule is in terms of EFPY of plant operation. Other factors that must be considered in establishing the surveillance capsule withdrawal schedule are the maximum fluence values at the vessel surface and 1/4 thickness (T) location.

Because the withdrawal schedule in Reference IV.1.C.3 is based on plant operation during the original 40-year license term, Reference 15 clarifies the requirements for 60-year license term by stating:

> *The plant-specific or integrated surveillance program shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation. The program withdraws one capsule at an outage in which the capsule receives a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation and tests the capsule in accordance with the requirements of ASTM E185-82.*

The surveillance capsule withdrawal schedule is generated based upon the guidelines specified in Reference IV.1.C.3, Section 7.6, and Reference IV.1.C.16. The surveillance capsule withdrawal schedules for Farley Units 1 and 2 are contained in the PTLR.

The current surveillance capsule withdrawal schedule for Farley Units 1 and 2 is based on Reference IV.1.C.3, which states the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal.

Because the fluence projections were revised to take into account the MUR-PU, a calculation of ΔRT_{NDT} at 54 EFPY was performed to determine if the MUR-PU fluences alter the number of capsules needed to be withdrawn for Farley Units 1 and 2 to satisfy the requirements of Reference IV.1.C.3 through EOLE (54 EFPY). This calculation of ΔRT_{NDT} is equal to the calculation of ΔRT_{PTS}. Therefore the ΔRT_{NDT} can be taken from Tables IV.1.C-1 and IV.1.C-2. For Unit 1, the largest shifts in RT_{NDT} are projected to be above 100°F, but below 200°F. According to Table 1 of Reference IV.1.C.3, four capsules are required to be pulled during the plant life. For Unit 2, the largest shifts in RT_{NDT} are projected to be above 200°F. According to Table 1 of Reference IV.1.C.3, five capsules are required to be pulled during the plant life.

All six surveillance capsules have been withdrawn and tested from each of the Farley Units 1 and 2 RVs. In order to satisfy the requirements of Reference IV.1.C.3, as clarified by Reference IV.1.C.16, the capsule data must encompass a fluence greater than the peak 60-year RV fluence, but no greater than twice this fluence.

The SER for Farley Unit 1 License Renewal (Reference IV.1.C.17) evaluates Capsule V for compliance with Reference IV.1.C.3 for a 60-year operating license. Capsule V was withdrawn and tested when the fluence on the capsule reached 7.22×10^{19} n/cm^2. This fluence is between one and two times the projected 60-year (54 EFPY) peak vessel fluence (6.05×10^{19} n/cm^2) as recommended in Reference IV.1.C.16. Therefore, Capsule V continues to satisfy the recommendations for a 60-year life under MUR-PU conditions.

The SER for the Farley Unit 2 License Renewal (Reference IV.1.C.17) evaluates Capsule Y for compliance with Reference IV.1.C.3 for a 60-year operating license. Capsule Y was withdrawn and tested when the fluence on the capsule reached 6.87×10^{19} n/cm^2. This fluence is between one and two times the projected 60-year (54 EFPY) peak vessel fluence (6.00×10^{19} n/cm^2) as recommended in Reference IV.1.C.16. Therefore, Capsule Y continues to satisfy the recommendations for a 60-year life under MUR-PU conditions.

The surveillance capsules withdrawn to date for Farley Units 1 and 2 meet the requirements of Reference IV.1.C.3 as required by Reference 1 IV.1.C.18 and satisfy the guidance contained in Reference IV.1.C.16. No change to the surveillance withdrawal schedule is required to support the MUR-PU. Thus, the previous capsule withdrawals remain valid. The Farley Units 1 and 2 surveillance program capsule fluence values are summarized in Tables IV.1.C-7 and IV.1.C-8, respectively, based on the updated fluences with the MUR-PU operating conditions.

Table IV.1.C-7: Farley Unit 1 Withdrawal Condition of the Surveillance Capsules

Capsule	Capsule Location (Symmetric Equivalent)	Withdrawn (EOC)	Withdrawal EFPY[1]	Capsule Fluence[1] (E > 1.0 MeV) (n/cm^2)	Maximum Vessel Fluence[1] (E > 1.0 MeV) (n/cm^2)	Lead Factor[1]
Y	343° (17°)	1	1.15	6.20E+18	1.98E+18	3.13
U	107° (17°)	4	3.09	1.76E+19	5.45E+18	3.23
X	287° (17°)	7	6.11	3.10E+19	9.59E+18	3.24
W	110° (20°)	12	12.42	4.80E+19	1.64E+19	2.92
V	290° (20°)	18	20.17	7.22E+19	2.46E+19	2.93
Z	340° (20°)	21	24.26	8.53E+19	2.92E+19	2.92

Note:
1. Updated as a part of the MUR-PU fluence evaluation.

Table IV.1.C-8: Farley Unit 2 Withdrawal Condition of the Surveillance Capsules

Capsule	Capsule Location (Symmetric Equivalent)	Withdrawn (EOC)	Withdrawal EFPY(1)	Capsule Fluence(1) (E > 1.0 MeV) (n/cm^2)	Maximum Vessel Fluence(1) (E > 1.0 MeV) (n/cm^2)	Lead Factor(1)
U	343° (17°)	1	1.11	6.13E+18	1.95E+18	3.15
W	110° (20°)	4	3.96	1.75E+19	6.39E+18	2.73
X	287° (17°)	6	6.43	3.02E+19	9.22E+18	3.27
Z	340° (20°)	12	13.86	4.97E+19	1.74E+19	2.86
Y	290° (20°)	16	19.01	6.87E+19	2.29E+19	3.01
V	107° (17°)	18	21.82	8.85E+19	2.56E+19	3.46

Note:
1. Updated as a part of the MUR-PU fluence evaluation.

References for Section IV.1.C

IV.1.C.1 Westinghouse Report, WCAP-17122-NP, Revision 0, "J. M. Farley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," October 2009.

IV.1.C.2 Westinghouse Report, WCAP-17123-NP, Revision 1, "J. M. Farley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," July 2011.

IV.1.C.3 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.

IV.1.C.4 10 CFR 50, Appendix G, "Fracture Toughness Requirements," Federal Register, dated December 12, 2013

IV.1.C.5 NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," U.S. Nuclear Regulatory Commission, October 2014. *[Agencywide Documents Access and Management System (ADAMS) Accession Number ML14149A165]*

IV.1.C.6 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, dated January 4, 2010.

IV.1.C.7 Westinghouse Report, WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR- M3G and FERRET," July 2018.

IV.1.C.8 United States Nuclear Regulatory Commission Regulatory Guide 1.190, Revision 0, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

IV.1.C.9 RSICC Data Library Collection DLC-185, "BUGLE-96: Coupled 47 Neutron, 20 Gamma-Ray Group Cross-Section Library Derived from ENDF/B-VI for LWR Shielding and Pressure Vessel Dosimetry Applications," July 1999. Available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratory.

IV.1.C.10 SNC Letter, "Joseph M. Farley Nuclear Plant, Unit 1 Revision 6 Pressure Temperature Limits Report, Unit 2 Revision 6 Pressure Temperature Limits Report," March 5, 2015. [ADAMS Accession Number ML15064A023]

IV.1.C.11 U. S. NRC, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. *[ADAMS Accession Number ML003740284]*

IV.1.C.12 Westinghouse Report, WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004.

IV.1.C.13 Westinghouse Letter, MCOE-LTR-13-17-P, Revision 0, "Farley Units 1 and 2 PTLR LAR NRC RAI Response," March 2013. (Westinghouse Proprietary).

IV.1.C.14 Westinghouse Letter, MCOE-LTR-13-17-NP, Revision 0, "Farley Units 1 and 2 PTLR LAR NRC RAI Response," March 2013.

IV.1.C.15 NRC Letter "Joseph M. Farley Nuclear Plant, Units 1 and 2, Issuance of Amendments Regarding Technical Specifications Revisions Associated with the Low Temperature Overpressure Protection System and the Pressure and Temperature Limits Report (TAC Nos. ME9256 and ME 9257) (NL-12-0868)," October 2, 2013. *[ADAMS Accession Number ML13249A386].*

IV.1.C.16 NRC Report, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," December 2010. *[ADAMS Accession Number ML103490041*]

IV.1.C.17 NRC Report, NUREG-1825, "Safety Evaluation Report Related to the License Renewal of the Joseph M. Farley Nuclear Plant, Units 1 and 2," Docket Nos. 50-348 and 50-364, May 2005. *[ADAMS Accession Number ML051250126]*

IV.1.C.18 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, dated January 31, 2008.

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:

Table IV.1.D-1 provides the ASME code of record for each component in the RCS, as shown in FSAR Table 3.2-4. There were no changes to the ASME code of record.

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

Component	Code	Code Class	Edition and Addenda
Piping and Supports	ASME Section III	1	1971 Edition, through the Summer 1971 Addenda
Steam Generator, tube side and shell side	ASME Section III	A	1989 Edition with no Addenda
Reactor Vessel	ASME Section III	A	1968 Edition through the Summer 1970 Addenda
Replacement Reactor Vessel Closure Heads	ASME Section III	1	1998 Edition through the 2000 Addenda
Reactor Coolant Pump Casing	ASME Section III	1	1971 Edition (Unit 1), and 1971 Edition with Addenda through Summer 1972 (Unit 2)
CRDMs	ASME Section III	A	1968 Edition through Winter 1969 Addenda
Pressurizer	ASME Section III	A	1968 Edition through Summer 1970 Addenda

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.

IV.1.E.i <u>In-service Inspection Program</u>

RESPONSE:

The In-Service Inspection (ISI) program is described in FSAR Subsection 5.2.8.6.2 and is in compliance with the requirements of 10 CFR 50.55a(g). The ISI program provides details for the examination of Class 1, 2, and 3 pressure-retaining items, their supports, and Class MC pressure-retaining items.

The ISI program plan states that system classifications are based on 10 CFR 50 and RG 1.26. The ISI plan boundary diagrams identify the boundary transitions between Class 1 and Class 2 or Class 3 components; Class 2 and Class 3 components; and between either Class 1, Class 2, or Class 3 and non-class components. It also includes sketches that provide the details required to specifically identify the location of components or the parts of components (e.g., welds or bolting) that require examination or testing. The sketches show the locations of supports or support structures requiring examination.

The MUR-PU does not affect the classifications or boundaries for ASME Class 1, 2, and 3 pressure retaining items, their supports, and Class MC pressure-retaining items. The MUR-PU does not affect the inspection requirements for ASME Class 1, 2, 3, and MC components and their supports as described in the ISI plan. The ISI program is implemented in accordance with 10 CFR 50.55a(g). The Farley program documents and the FSAR each conform to the requirements of 10 CFR 50.55a(g) and are unaffected by the MUR-PU. The Farley TSs do not directly refer to the ISI program. Therefore, the ISI program will continue to comply with the licensing bases of 10 CFR 50.55a(g).

IV.1.E.ii In-service Testing Program

RESPONSE:

The In-service Testing (IST) program is described in FSAR Subsection 5.2.8.6.3 and is in compliance with the requirements of 10 CFR 50.55a(f). The section addresses the applicability of the IST programs for both Units. The scope of the IST program is to establish IST requirements for the following safety-related Class 1, 2, and 3 pumps, valves, and pressure relief devices that are required in shutting down the reactor to the safe shutdown condition (hot shutdown for Farley), maintaining the safe shutdown condition, or mitigating the consequences of an accident:

- Pumps with an emergency power source required to perform the above
- Active or passive valves required to perform the above
- Pressure relief devices protecting systems or portions of systems that are required to perform the above

The Farley program documents, FSAR, and TSs all conform to the requirements of 10 CFR 50.55a(f) and are unaffected by the MUR-PU. Therefore, the IST program will continue to comply with the licensing bases of 10 CFR 50.55a(f).

IV.1.E.iii Flow-Accelerated Corrosion Program

RESPONSE:

Flow-accelerated corrosion (FAC) is a form of material degradation that, under certain flow and chemistry conditions, results in thinning of the inside pipe wall in carbon steel piping and fittings. Similarly, erosion causes wall thinning, leaks, and ruptures. The Farley FAC program is administered by SNC Procedure NMP-ES-011 and its sub-tier documents. The program was developed to be in compliance with NRC Generic Letter (GL) 89-08 and to be consistent with

the ten attributes of the aging management program described in NUREG-1801, Section XI.M17.

Farley uses the CHECWORKS computer program to model piping systems for possible FAC degradation. Plant design and operating data, together with a predictive algorithm, are used to predict the rate of wall thinning and remaining service life on components in the systems. The resulting predictions are then used as inputs to the plant's FAC program to help select inspection locations.

True North Consulting was contracted to update Farley's FAC model to include changes to the heat balance for MUR-PU conditions. True North issued Technical Reports BP-2019-0001-01-TR for Unit 1 and BP-2019-0001-02-TR for Unit 2. These reports provide comparisons of pre- and post-uprate wear rates, based on the MUR-PU power level, to be used as input to the FAC program. The purpose of the True North evaluation for each Unit was to update the Farley CHECWORKS model to include the changes to the heat balance due to the MUR-PU. The models were then used for the following:

1. Generate relative FAC wear rate rankings for piping components.
2. Generate FAC wear rate predictions calibrated to the inspection data.
3. Predict remaining life based upon predicted FAC wear rates.

The CHECWORKS model results included a water chemistry analysis performed on every water chemistry period to ensure that chemistry values were reasonable. The water chemistry analysis uses the plant global data (heat balance diagram [HBD], power level data, steam cycle data, water chemistry data, and plant period data) to determine the chemistry conditions at various locations around the steam cycle. These values strongly affect FAC wear rates. Water chemistry analysis calculates pH, dissolved oxygen concentration, constituent concentration, and hydrazine concentration at each location on the CHECWORKS HBD. The appropriate values are then used in the calculation of predicted wear rates for each component through the association of the line to the HBD.

The results of each report were comparisons of wear rates between pre-MUR-PU and post-MUR-PU operating conditions. Some rates increased, while others decreased. The data was then used as inputs into the Farley FAC program. The selection of locations for FAC examination is based on a variety of criteria, including the wear rate data from the CHECWORKS model. The results from the True North models are, therefore, fed into the FAC program for consideration of inspection locations and may or may not alter those determinations. However, the wear rate data do not, in themselves, affect the FAC program. Additionally, the FAC program specifies that all inspection data taken during maintenance/refueling outages be input to CHECWORKS during or following the outage. As such, the CHECWORKS model is updated after every refueling outage with inspection information, plant conditions, and water treatment data (dissolved oxygen and pH) from the chemistry department. This requirement is not affected by the MUR-PU.

The revised CHECWORKS models provided by True North Consulting result in some components' wear rates increasing due to the MUR-PU. These increased wear rates may alter decisions as to the selection of locations for examination. However, the FAC program itself is not affected by the MUR-PU.

In conclusion, the FAC program documents and the FSAR were reviewed and found to be unaffected by the MUR-PU. The FAC program will continue to comply with the licensing bases of NRC GL 89-08 and to be consistent with the 10 attributes of the aging management program described in NUREG-1801, Section XI.M17.

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 concern was based on a tube fatigue incident that occurred at the North Anna plant in 1987. From that incident, the main contributing factors identified were the existence of:

- Carbon steel support plates with occurrence of corrosion products and tube denting
- Lack of AVB support for the affected tubing
- Fluid-elastic instability in the region of the tube

In the case of the Farley Units 1 and 2 Model 54F RSGs, the design has utilized stainless steel support plates which preclude the tube denting mechanism from occurring. Also as explained in section IV.1.A.vi.c, high cycle fatigue is not predicted for tubing with the design basis support plates and no tube denting. Therefore, the fatigue issue of NRC Bulletin 88-02 has been addressed and is not a concern for these RSGs as well as at the MUR-PU conditions.

References for Section IV:

See the individual reference listings for Sections IV.1.A and 1V.1.C.

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. 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:

The original license application for Farley Units 1 and 2 was for 2,652 MWt. Subsequently, SNC undertook a program in 1998 to uprate Farley Units 1 and 2 to a maximum reactor core power level of 2,775 MWt. The MUR-PU will increase the core power level to 2821 MWt. The impact of this increase in power on the power block electrical equipment was evaluated at a conservative NSSS power level of 2841 Mwt as follows.

Main Generator

The nameplate rating of each main generator (MG) for Farley Units 1 and 2 is 1,045 MVA (based on 75 psig hydrogen pressure), 0.85 power factor, 22 kV, three-phase, 60 Hz at 1800 rpm. This increase in thermal power results in an increase in the generator electrical power output as follows:

- Unit 1:
  - 944.7MW, 1045MVA, 447MVAR, at 0.90 lagging power factor
  - 944.7MW, 971MVA, 225MVAR, at 0.97 leading power factor
- Unit 2:
  - 953.3MW, 1045MVA, 428MVAR, at 0.91 lagging power factor
  - 953.3MW, 978MVA, 220MVAR, at 0.97 leading power factor

The evaluation of the MGs concluded that, when the plant is operating at the MUR-PU level, the MG for each of Units 1 and 2 shall be capable of performing the assigned functions without exceeding equipment ratings. Subsystems and components shall be capable of performing the design functions without exceeding the limiting design parameters of the individual components. The MG will be operated within its generator capability curve at MUR-PU conditions.

Therefore, the increased output from the MUR-PU remains within the MG maximum capability curves.

Main Transformers

The main transformer (MT) bank for each of Units 1 and 2 consists of three single-phase transformers. The MT bank has a maximum design rating of 1200 MVA, which is greater than the MG output capability of 1045 MVA. Therefore, the MTs are sized adequately for the maximum MUR-PU conditions.

The evaluation determined that there are no changes to the protective device settings, or cable sizing; and there are no changes to the current AOR for Units 1 and 2.

Unit Auxiliary Transformers

Power required for Units 1 and 2 station auxiliaries during normal operation is supplied via Unit Auxiliary Transformers (UATs) 1B and 2B. Unit Auxiliary Transformers 1A and 2A are available onsite, but they are normally disconnected from the plant. The UATs supply power to the units' 4.16-kV medium voltage buses. The only 4160-VAC loads affected by the uprate are Condensate Pumps 1A, 1B, and 1C for Unit 1, and Condensate Pumps 2A, 2B, and 2C for Unit 2. The brake horsepower (BHP) of the condensate pump for the MUR-PU condition will increase by 25 HP but remains within the 3,000-hp, 1.15 Service Factor rating of the motor.

The evaluation of the impact of the MUR-PU on the UATs determined that the load changes at MUR-PU conditions are within the transformer design ratings. The evaluation determined that there are no changes to the protective device settings or cable sizing.

Start-Up Auxiliary Transformers

The start-up auxiliary transformers (SATs) supply offsite power from the switchyard. The SATs' electrical loads, which are affected by the MUR-PU, are the same as the UATs. No measurable load increases on the 4.16-kV medium voltage buses have occurred as a result of the MUR-PU. The only 4160-VAC loads affected by the uprate are Condensate Pumps 1A, 1B, and 1C for Unit 1, and Condensate Pumps 2A, 2B, and 2C for Unit 2. The BHP of the condensate pump for the MUR-PU condition will increase by 25 HP but remains within the 3,000-hp, 1.15 Service Factor rating of the motor.

The evaluation demonstrates that the SAT load changes at MUR-PU are within the transformer ratings. The rating of each SAT will accommodate its load under MUR-PU conditions. There are no changes to the ratings of the SATs, no changes to the protective device settings, or cable sizing.

Switchyard

The Farley Units 1 and 2 switchyards are owned and maintained by Alabama Power. The 230-kV and 500-kV switchyards both employ a breaker-and-a-half arrangement to provide the necessary operating flexibility and, consequently, reliability.

The connection between the Unit 1 MT 230-kV high voltage and the first circuit breaker in the switchyard has been designed to accommodate a maximum output current of the MT based on the Unit 1 MT's 1200 MVA rating. The connection between the Unit 2 MT 500-kV high voltage and the first circuit breaker in the switchyard has been designed to accommodate a maximum output current of the MT based on the Unit 2 MT's 1200 MVA rating. The evaluation has determined that, when the plant is operating at the MUR-PU power level, the switchyard can accommodate the increase in the output current of the MT in Units 1 and 2.

The connection between the start-up transformer and the switchyard is not impacted, since the loads on the SATs are not impacted.

DC Power System

The direct current (DC) power system is designed to provide a reliable source of continuous power for control, instrumentation, and emergency lighting and redundant subsystems. The DC

power system is comprised of safety-related systems and non-safety-related systems, as described in the following sections.

The evaluation of the MUR-PU on the DC power system determined that there will be no change in the electric power requirements imposed as a result of operation under the MUR-PU. When each of Units 1 and 2 is operating at the MUR-PU level, the DC power systems shall be capable of performing their assigned functions without exceeding equipment ratings or industry guidelines. Subsystems and components are capable of performing their design functions without exceeding the limiting parameters of the individual components.

Alternating Current System

The AC auxiliary system for Farley Units 1 and 2 consists of the medium voltage 4.16-kV system and the low voltage 600-V, 480-V and 208-V systems.

The Unit 1 AC auxiliary system receives power under normal operating conditions from the Unit 1 MG via UAT 1B. Under start-up, normal, and shutdown conditions, power is supplied from Start-Up Transformer 1B.

The Unit 2 AC auxiliary system receives power under normal operating conditions from the Unit 2 MG via UAT 2B. Under start-up, normal, and shutdown conditions, power is supplied from Start-Up Transformer 2B.

In both units, medium voltage power is supplied from the UATs and the SATs to the 4.16-kV system and from this system to the low voltage systems. The only 4160 VAC loads affected by the uprate are the Condensate Pumps. The BHP of the Condensate Pump for the MUR condition will increase by 25 HP, but remains within the 3,000HP, 1.15 Service Factor rating of the motor. There are no other increases in the electrical loads in the 4160 VAC system.

The evaluation of the AC system determined that there are no changes to the 4160-V bus ratings, protective device settings, or cable sizing; and there are no changes to the current AOR for Units 1 and 2.

Low Voltage AC Power System

The low voltage AC (LVAC) power system consists of the 120-VAC vital power, 120-VAC regulated power, and 208/120-VAC power systems for Farley Units 1 and 2. The 120-VAC vital power that supplies power to the LEFMs is not from safety-related buses. The load of the LEFMs has no impact on safety-related motors or the condensate pump motors. A maximum load of 5A is expected for each set of Caldon LEFMs. This increase in loading is bounded within the existing AOR for each unit.

The evaluation of the LVAC power system determined that when the plant is operating at the MUR-PU level, the LVAC power system shall be capable of performing its assigned functions without exceeding equipment ratings or industry guidelines.

V.1.A Emergency Diesel Generators

RESPONSE:

The onsite emergency AC power supply for Farley Units 1 and 2 consists of five diesel generators that supply standby power for 4160-V emergency buses of each Unit when offsite power is unavailable.

The loading on the emergency diesel generators was evaluated to determine potential changes at the MUR-PU. A review of the electrical loads for operation at MUR-PU conditions indicates that there are no load additions or modifications, and no changes to load sequences or durations required to the existing emergency diesel generators. Therefore, there is no impact to the existing emergency diesel generator loading analysis, which bounds the MUR-PU conditions. The emergency onsite power system has adequate capacity and capability to provide onsite standby power for safety-related loads following a LOSP. Therefore, the existing emergency onsite power system bounds the design requirements at the MUR-PU.

V.1.B Station blackout equipment

RESPONSE:

The licensing bases for station blackout (SBO) have been reviewed for potential impact from a MUR-PU. The licensing bases include the capability to provide core cooling, the ability to power Class 1E battery chargers, the ability to provide a source of compressed air, the ability to cope with a loss of ventilation, the ability to maintain appropriate containment integrity, and an ability to maintain adequate RCS inventory. Evaluation of the SBO licensing bases revealed that the licensing bases for SBO continue to be met under MUR-PU conditions.

<u>Station Blackout Duration</u>

For the SBO duration, the plant is required by 10 CFR 50.63 to be capable of maintaining core cooling and appropriate containment integrity. The minimum acceptable SBO coping duration for Farley Units 1 and 2 is four hours. The coping duration was based on offsite power design characteristics, emergency AC power configuration group, and targeted emergency diesel generator reliability. None of the factors that went into deciding the SBO coping duration for Farley Units 1 and 2 are affected by the implementation of the MUR-PU. Therefore, the licensing basis coping duration of 4 hours is also unaffected by the implementation of the MUR-PU.

<u>Alternate AC Power Source</u>

Farley Units 1 and 2 selected the alternate AC (AAC) approach for coping with an SBO event and dedicated Class 1E diesel Generator 2C as the AAC power source to cope with an SBO event in either Unit for the required duration. This configuration will be unaffected by the implementation of the MUR-PU. Therefore, the licensing basis for crediting the use of diesel generator 2C as the AAC power source is unaffected by the implementation of the MUR-PU.

Station Blackout Coping Capability

Condensate Inventory for Decay Heat Removal

The condensate storage tank (CST) volume requirements in TS 3.7.6 envelope the required storage capacity for dedicated safety grade water for an SBO at MUR-PU conditions. The current minimum required useable inventory is based on a reactor trip from 102 percent of the current power level of 2,775 MWt which accounts for the proposed MUR-PU. Therefore, the licensing basis for decay heat removal during an SBO will continue to be met.

Class 1E Battery Capacity

Station battery chargers aligned to the station's Class 1E batteries are powered by the station's credited AAC source, which will be available within ten minutes. There are no required changes to the station's DC power or emergency onsite AC power supply/standby power systems for operation at MUR-PU conditions. Therefore, the MUR-PU will have no effect on the ability to power the station's battery chargers during an SBO at MUR-PU operation, and the licensing basis for Class 1E battery availability during an SBO will continue to be met.

Compressed Air

Compressed air can be manually aligned to the AAC source to supply air-operated valves necessary for safe shutdown. Therefore, an air compressor is connectable to an engineered safety feature (ESF) bus that is powered by the AAC source and is available during an SBO event. The station's credited AAC source will be available within ten minutes. There are no required changes to the compressed air or emergency onsite AC power supply/standby power systems for operation at MUR-PU conditions. Therefore, the MUR-PU will have no effect on the ability to provide a source of compressed air during an SBO, and the licensing basis for air compressor availability during an SBO will continue to be met.

Effects of Loss of Ventilation

Heating, ventilation, and air conditioning (HVAC) systems relied upon to provide ventilation during an SBO are powered by the station's credited AAC source, which will be available within ten minutes. There are no required changes to the station's HVAC or emergency onsite AC power supply/standby power systems for operation at MUR-PU conditions. Therefore, the MUR-PU has no impact on the ability to cope with the effects of a loss of ventilation during an SBO, and the licensing basis for coping with a loss of ventilation during an SBO will continue to be met.

Containment Isolation

Valves identified as containment isolation valves of concern for SBO that would be required to be operable during an SBO event are not adversely affected by the MUR-PU. The station's credited AAC source, which will be available within ten minutes, powers one division of safety buses in order to maintain containment integrity. There are no required changes to the station's AC power or emergency onsite AC power supply/standby power systems for operation at MUR-PU conditions. Therefore, the MUR-PU has no impact on the ability to maintain containment isolation during an SBO, and the licensing basis for maintaining containment integrity during an SBO will continue to be met.

Reactor Coolant Inventory

Pumps from the chemical and volume control system (CVCS) are powered by the station's credited AAC source, which will be available within ten minutes. Each pump has a capacity that exceeds the assumed RCS leak rate of 100 gpm during an SBO event. There are no required changes to the station's CVCS or emergency onsite AC power supply/standby power systems for operation at MUR-PU conditions. Therefore, the MUR-PU has no impact on the requirements of maintaining adequate RCS inventory during an SBO.

V.1.C Environmental qualification of electrical equipment

RESPONSE:

The effect on the Farley equipment qualification analysis as a result of implementing the MUR-PU has been evaluated. This evaluation determined the following:

1. The equipment and components of the equipment qualification program continue to operate satisfactorily and perform their intended functions at the uprated conditions to satisfy the requirements outlined in 10 CFR 50.49, and the safety-related electrical equipment is qualified to survive the environment at its specific location during normal operation and during an accident.
2. The equipment qualification program equipment accommodates MUR-PU conditions without exceeding electrical equipment qualification margins for the parameters of temperature, pressure, radiation, and similar parameters, as defined by IEEE Standard 323-1974.

An evaluation to determine the effect, if any, on the Farley nuclear equipment qualification analysis as a result of implementing the MUR-PU was performed. The following conditions were evaluated:

- Containment pressure and temperature analyses
- Containment flooding
- MSLB in the MSVR
- Other HELBs pressure / temperature outside containment
- Post-LOCA sump water pH
- Radiation environments to support equipment qualification

The evaluations determine that there is no impact on the existing analyses or changes to equipment qualification areas and, therefore, the existing analyses remain bounding and the MUR-PU will not affect equipment in the equipment qualification program for EQ.

V.1.D Grid Stability / Transmission Planning Study

RESPONSE:

The following studies have been conducted starting with load flow base cases developed for use as one possible scenario at the future that could occur. These base cases incorporate the most recent information available concerning loads, transmission enhancements, committed

generation, firm transactions, and other information pertinent to building load flow cases. For later years, they also incorporate assumptions on transmission and generation additions that may not be budgeted and or which commitments may not have been made by the entities represented in the models. Information from other transmission owners within the Southern Balancing Authority Area (SBAA) was obtained and incorporated in these models. The systems external to the SBAA were obtained either from the given balancing authority or from the most recent NERC Multiregional Modeling Working Group cases.

Note: "Designation Request" shall hereafter refer to the 48 MW incremental designation requested from the Farley facility to serve the native load of Southern Companies.

Local Area System Impacts

Two scenarios were developed as follows:

1. Study "Off" Case: Base case with prior queued requests, potential rollover, and other queuing sensitivities considered.
2. Study "On" Case: Study "Off" case with the Designation Request modeled in effect.

A steady state analysis did not identify any transmission constraints that require transmission projects attributable to the Designation Request.

Stability Impacts

A stability analysis did not identify any stability-related transmission constraints attributable to the Designation Request. The expected clearing times are less than or equal to the critical clearing times.

The breakers adjacent to a 230 kV line at Farley should continue to be independent pole-operated rather than gang-operated. The breakers in the 230 kV switchyard that are not adjacent to a line can continue to be gang-operated.

Nuclear Plant Offsite Power Impacts

A review of nuclear plant offsite power requirements did not identify any transmission projects attributable to the Designation Request to ensure bus voltages remain within the acceptable ranges.

Bus Ampacity Impacts

Bus ampacity analysis did not identify any transmission constraints that require transmission projects attributable to the Designation Request.

Interface Transfer Capability Impacts

An assessment of the impact of the Designation Request on import and export transfer capability across the SBAA interlaces was performed. The impact analysis did not identify any transmission constraints that require transmission projects attributable to the designation request.

References for Section V:

None

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:

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

VI.1.A.i Main Steam

RESPONSE:

The main steam supply system (MSSS) is described in FSAR Section 10.3; and the turbine bypass system (TBS) is described in FSAR Section 10.4.4. The MSSS carries steam from the three SGs to the turbine generator, the MSRs, the steam jet air ejector system, the turbine shaft gland seals, the SG feedwater pump turbines, the turbine-driven AFW pump, and the TBS. The TBS bypasses main steam directly to the main condenser during the transient conditions of a sudden load rejection by the turbine generator or a TT and during plant startup and shutdown.

An evaluation of the MSSS piping at MUR-PU conditions was performed. As MSSS pressures and temperatures at MUR-PU conditions are bounded by piping design parameters, the MSSS piping is acceptable at MUR-PU conditions.

An evaluation of the impact of the MUR-PU was performed for the MSIVs. The MSIVs are designed to close within a set time to prevent a SG from blowing down due to an MSLB or to preclude overpressurization of containment due to reverse flow from the other main steam lines if there is a break inside containment. The ability of the MSIVs to close within the required time is not affected by the MUR-PU because valve construction is such that increased uprate steam flow and pressure drop across the valve act to assist in closing the valve. The MUR-PU does not alter the physical arrangement of the MSS or any of the MSIVs. Therefore, the MUR-PU does not affect the ability of the MSIVs to close and prevent an overpressure condition inside the containment building. Additionally, the MSIVs are designed to fail closed if the instrument air-operated actuator holding the valve open loses air pressure, since the actuator contains a spring that forces the valve closed. The MUR-PU does not change the MSIV actuator or the air supplied to it in any way. Therefore, the MSIVs' fail-closed function is not affected by the MUR-PU. The MSIVs are capable of accommodating the increased steam flow rate at MUR-PU conditions without modifications. As a result, the MSIVs are acceptable for MUR-PU conditions.

The MSIV bypass valves are provided to permit equalizing steam pressure across the MSIVs before reopening after a trip, or to warm the main steam lines during startup. The MSIV bypass valves are not affected by the MUR-PU, since they are only used during low flow steam

conditions. The maximum closing time is also not affected by the MUR-PU, since the low flow operating conditions do not change. Therefore, the MSIV bypass valves are acceptable for MUR-PU conditions.

The MSSVs were evaluated at MUR-PU conditions. The MSSVs are designed to protect the SGs and the MSSS from overpressurization conditions and their set pressures are staggered to prevent chattering during operation. The MSSVs are capable of passing the increased steam flow rate at MUR-PU conditions and are capable of meeting the maximum capacity criterion for MUR-PU conditions without modifications. The design pressure of the SGs does not change for the MUR-PU. Thus, the set points of the MSSVs also do not change. As a result, the MSSVs are acceptable for MUR-PU conditions.

There is one atmospheric relief valve (ARV) on each main steam line located outside containment upstream of the MSIVs and its primary function is to provide a means of decay heat removal when the main heat sinks are not available. The total relieving capacities of the ARVs are sufficient to permit cooling the plant down from zero load hot standby temperature to the RHR system initiation temperature at an average cooldown rate of 50°F per hour, beginning at two hours following a reactor trip at MUR-PU conditions. Therefore, the ARVs are acceptable for MUR-PU conditions.

The TBS provides an artificial load by dumping excess steam to the condenser via the eight steam dump valves (SDVs). Steam dump in conjunction with the reactor control system permits the NSSS to withstand an external load reduction of up to 50 percent of plant-rated electrical load without a reactor trip. The NSSS control systems margin-to-trip analysis confirmed the steam dump system capability at MUR-PU conditions. There is acceptable margin to the relevant reactor trip setpoints during and following the 50-percent load rejection transient. The SDV maximum closing time of five seconds and opening time of three seconds are not affected by MUR-PU conditions. Therefore, the TBS is adequate for MUR-PU conditions.

In conclusion, the MSSS and TBS systems were evaluated to determine the impact of the MUR-PU. MUR-PU operating parameters are bound by the original piping, component, and equipment design parameters, or by the original design considerations for off-normal operation. Therefore, the MSSS and TBS are acceptable at MUR-PU conditions.

VI.1.A.ii Main Turbine-Generator

RESPONSE:

As discussed in FSAR Sections 10.1 and 10.2, the turbine-generator converts the thermal energy of the steam generated by the NSSS into electrical energy. The turbine is a Westinghouse (modified by Siemens) 1800 rpm, tandem compound, 4-flow exhaust with 45.5-in. last-stage blades. It is equipped with an automatic stop and emergency trip system which will trip the stop and control valves to a closed position in the event of turbine overspeed, low bearing oil pressure, low vacuum, or thrust bearing failure. The MG system output is rated for 1,045 MVA, 22kV, and 27,424A for both Units.

The main turbine was reviewed for MUR-PU conditions and it was determined that high pressure blading design (specifically the first and second rotating blade rows on each end) will require modernization for MUR conditions. Additionally, the existing Units 1 and 2 high pressure

turbines require modernization to increase valve wide-open steam flow capacity, and to recover turbine throttle flow margin to support the MUR-PU. The existing BB296 high pressure turbine internal components will be replaced with BB296FG high pressure modernizations as part of the Unit 1 and 2 Design Change Packages in accordance with the Farley engineering design change process.

The MG system output is rated for 1,045 MVA, 22kV, and 27,424A for both Units. Per the Turbine analyses work performed, the maximum output will increase from 921.5 megawatts (MW) to 944.7 MW which is an increase of 23.2 MW for Unit 1, and from 931 MW to 953.3 MW which is an increase of 22.3 MW for Unit 2. This increase can be accommodated by the 1045MVA rating of the generator, provided it is operated within the power factors and MVAR limits below:

- Unit 1:
    - 944.7MW, 1045MVA, 447MVAR, at 0.90 lagging power factor
    - 944.7MW, 971MVA, 225MVAR, at 0.97 leading power factor
- Unit 2:
    - 953.3MW, 1045MVA, 428MVAR, at 0.91 lagging power factor
    - 953.3MW, 978MVA, 220MVAR, at 0.97 leading power factor

Therefore, the generators for both units are acceptable for use after the high pressure turbine upgrade and the MUR uprate.

There are no other changes required to the main turbine to support the MUR-PU.

The high pressure turbine replacement does not require an update to the turbine missile analysis. Low-pressure turbine missiles have a higher probability and are the most likely source of a missile generated by the turbine. This is due, in part, to failure mechanisms for high-pressure turbines having large design margins. The FSAR only discusses the LP turbine missile and does not postulate an high pressure turbine missile. Additionally, internal failure of the high pressure rotor would be contained by the high strength high-pressure turbine casing. For these reasons, no turbine missile analysis update is required, and the existing turbine missile analysis is adequate for the MUR-PU.

VI.1.A.iii <u>Condensate and Feedwater</u>

RESPONSE:

The condensate and feedwater systems are described in FSAR Section 10.4.7.

The condensate system consists of three parallel, 50-percent capacity condensate pumps. Normally two condensate pumps are operating at full load delivering water to the feedwater pumps' suction header. The MUR-PU results in increased condensate flow of approximately 1.8 percent. Adequate condensate pump net positive suction head is available at uprate conditions. Piping pressures and temperatures are not significantly impacted by the MUR PU. Relevant parameter changes resulting from the power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the condensate system is acceptable at MUR-PU conditions.

The feedwater systems consist of two 50-percent, variable speed, turbine-driven feedwater pumps. Feedwater flow is controlled by the feedwater control valves on the pump discharge. A speed controller automatically controls the speed of the turbine-driven feedwater pumps. Inputs to the speed controller are steam flow, main steam header pressure, and feedwater pump discharge header pressure. The MUR-PU requires an adjustment to the main steam header/feedwater header differential pressure input setpoint. With the pump driver speed controller setpoint adjustment and the resulting increase in the pump speed, the turbine-driven feedwater pumps will provide the necessary flow in support of the MUR-PU. The changes result in the feedwater control valves' position remaining essentially unchanged from pre-MUR-PU operation. The MUR-PU results in increased feedwater flow of approximately 2.0 percent. Adequate feedwater pump net positive suction head is available at MUR-PU conditions, and no changes are required to the low suction pressure alarm and trip setpoints.

Feedwater isolation valves, feedwater control valves, feedwater control bypass valves, and feedwater pump discharge valves will continue to provide containment isolation capability. The existing NSSS accident analysis was completed at 102 percent of the current licensed core power, which bounds the MUR-PU. Piping pressures and temperatures are not significantly impacted. Relevant parameter changes resulting from the MUR-PU do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the feedwater system is acceptable at MUR-PU conditions.

The 50-percent load rejection transient was evaluated for MUR-PU conditions. There is no significant impact on system operation from this postulated transient.

There are two parallel trains of FWHs. Each train consists of five FWHs located on the suction side of the feedwater pumps and one heater on the discharge side. The thermal performance of all FWHs will not change significantly from the current operating condition to MUR-PU conditions. FWH shell and tube side operating pressures and temperatures at MUR-PU conditions remain below the design values of these parameters. Relevant FWH parameter changes resulting from the MUR-PU do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Therefore, the FWHs are acceptable at MUR-PU conditions.

VI.1.A.iv **Auxiliary Feedwater**

RESPONSE:

The AFWS is described in FSAR Section 6.5.

The AFWS serves as a backup system for supplying feedwater to the SGs when the main feedwater system is not available. The system includes two motor-driven pumps and one turbine-driven pump configured into three trains. Each pump takes suction from the CST.

The AFW system is designed as an engineered safety feature to provide redundant means of removing decay and sensible heat from the RCS via the SGs during emergency conditions. The system design is based on providing sufficient flow to prevent the loss of pressurizer vapor space during a feedwater line break with LOSP. The turbine-driven pump is designed to operate with steam produced in the SGs and deliver sufficient feedwater flow to safely cool down the RCS. No AC power is required for 2 hours of operation of the turbine-driven AFW pump. The

volume that defines the basis of operability in the TSs is based on maintaining the plant at hot standby conditions for 9 hours with steam discharge to the atmosphere concurrent with a LOSP.

The current AOR assumes a reactor trip occurs at 102-percent power. Therefore, the current AOR remains conservative and bounds the MUR-PU power level. The TS minimum CST volume requirement of 164,000 gallons ensures that the usable volume bounds the minimum CST volume requirements.

No physical changes are being made to the AFW system, and no physical or administrative changes are being made to the CST and required capacity. In addition, no changes are proposed for the design or operation of the AFW system pumps. Evaluations of limiting transients and accidents have determined that minimum AFW flow rates are acceptable for the MUR-PU. Therefore, the licensing bases of the AFW system will continue to be met at MUR-PU conditions.

VI.1.B Containment Systems

VI.1.B.i Containment Spray System

RESPONSE:

The containment spray system (CSS) is described in FSAR Section 6.2.2. The CSS operates to limit peak containment pressure to less than the design pressure of 54 psig during a LOCA or MSLB in order to maintain containment structural integrity. The CSS provides a cooling spray into the containment to remove heat from the containment atmosphere. Trisodium phosphate (TSP) dissolves into sump water from the baskets and is delivered to the containment spray (CS) rings during recirculation mode. During recirculation mode, the CSS takes water from the containment sump and delivers the discharge through 360-degree containment recirculation spray rings. The existing containment response analyses remain bounding for the MUR-PU. The CSS operating and design parameters in the FSAR bound the MUR-PU parameters. There are no new operating requirements imposed on the CSS as a result of the MUR-PU. Therefore, the CSS is acceptable for MUR-PU.

VI.1.B.ii Containment Air Cooling System

RESPONSE:

The containment air cooling system is described in FSAR Section 6.2.3.

As the post-accident containment atmosphere, which consists of a steam-air mixture, is circulated through the bank of cooling coils; it is cooled, and a portion of the steam is condensed. The capacity of one cooler in conjunction with one CS train is adequate to maintain pressure and temperature post-accident.

The system design basis is to provide adequate containment cooling from the operation of one train of CS and one containment cooler. The system limits the pressure transient inside the containment following a LOCA.

The containment air cooling system was analyzed at the MUR-PU conditions and was found to be satisfactory.

VI.1.B.iii ECCS Recirculation Sump pH Control System

RESPONSE:

In accordance with regulatory guidance provided in RG 1.183, Appendix A, the pH of the aqueous solution collected in the containment sump after completion of the injection mode of the CSS and the ECCS, and addition of all additives for reactivity control, fission product removal, and other purposes, should be maintained at a level sufficiently high to provide assurance that significant long-term iodine re-evolution does not occur. In accordance with RG 1.183, long-term iodine retention may be assumed only when the equilibrium sump solution pH is 7 or greater. Evaluations of pH should consider the effect of acids and bases created during the LOCA event (e.g., radiolysis products).

As noted in FSAR Subsection 6.2.3.4.1, chemical control of the containment sump water following a LOCA is achieved by the use of TSP contained in three baskets located in the recirculation sump area of the containment. Each basket has level marks which indicate the acceptable range of TSP volume for the basket which would raise the pH of the recirculating solution into the range of 7.0 to 10.5. The TSP will start to dissolve when the post-LOCA water level in the containment sump reaches the bottom of the TSP baskets. The TSP combines with the sump water in an equilibrium reaction and helps to maintain the sump pH in the required range.

Maintaining long term sump water pH in the required band is necessary to keep the radioactive iodine in solution and prevent re-evolution in support of dose consequences and to minimize chloride induced stress corrosion cracking of austenitic stainless steel components (pH ≥ 7.0), and corrosion control in support of equipment qualification (pH ≤ 10.5).

The MUR-PU will not affect the current masses of the RCS, accumulators, and the RWST as specified in the plant TSs and TRM.

In addition, the current long-term LOCA M&E releases remain applicable for MUR-PU operations, thus the containment sump water temperature transient is not affected by the MUR-PU.

Consequently, the current minimum sump pH value of 7.21 remains applicable for the MUR-PU and will remain within the required range of 7.0 – 10.5.

In conclusion the current sump pH estimates will remain valid for the MUR-PU.

VI.1.B.iv Containment Isolation

RESPONSE:

Containment isolation is described in FSAR Subsection 6.2.4.1.

Because the MUR power uprate does not change any of the accident analyses discussed in Section II, the existing design and operation for containment isolation remains the same. The containment isolation was reviewed at the MUR-PU conditions and was found to be satisfactory.

VI.1.C Safety related cooling water systems

VI.1.C.i Component cooling system

RESPONSE:

The CCWS is described in FSAR Section 9.2.2. The CCWS is a closed loop cooling system that transfers heat from reactor auxiliaries to the service water system (SWS) during plant operation and during normal and emergency shutdown / cooldown. The CCWS consists of three motor-driven cooling water pumps; three heat exchangers; a surge tank; associated piping, valves, instrumentation, and controls; and auxiliary electrical equipment. During normal operation, one reactor plant component cooling pump and one reactor plant component cooling heat exchanger accommodate the heat removal load. The CCWS is designed to provide the cooling requirements for normal plant operation, plant cooldown, SFP cooling, and DBA cooldown.

The CCWS was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the MUR-PU heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. The analysis confirms that, at MUR-PU conditions, normal plant operation and required cooldown time continue to be met. Therefore, the design and licensing bases of the CCWS will continue to be met at MUR-PU conditions.

VI.1.C.ii Nuclear service water system

RESPONSE:

The SWS is described in FSAR Section 9.2.1. The SWS consists of two redundant flow paths, each consisting of two service water pumps, a standby pump, two service water strainers, piping, and valves. The SWS provides cooling water for heat removal from the reactor plant auxiliary systems during all modes of operation and from the turbine plant auxiliary systems during normal operation. Each component cooled by the SWS was evaluated to confirm that the existing flow rate is sufficient to satisfy the power uprate heat removal requirements during normal, shutdown, and accident conditions. The evaluations determined that the existing SWS flows will continue to support the heat removal requirements at uprate conditions. The SWS and component design parameters remain bounding for power uprate operation. No system modifications are required to support the power uprate.

Therefore, the design and licensing bases of the SWS will continue to be met at MUR-PU conditions.

VI.1.C.iii Ultimate heat sink

RESPONSE:

The ultimate heat sink (UHS) is described in FSAR Section 9.2.5. The UHS for Farley Units 1 and 2 is the storage pond. The UHS is capable of providing sufficient cooling for at least 30 days to permit simultaneous safe shutdown and cooldown of both nuclear reactor units and to maintain them in a safe shutdown condition or, in the event of an accident in one unit, to permit safe control of the accident and simultaneously permit safe shutdown and cooldown of the other unit and maintain it in a safe shutdown condition. Sensible heat removed from both safety and

nonsafety-related cooling systems during normal operation, shutdown, and accident conditions is discharged via the service water and circulating water systems to the Chattahoochee River. Post accident evaluations assume no makeup is available from the river water system for a period of 30 days while the service water system is aligned to the UHS in the recirculation mode. The increased heat loads from MUR-PU remain bounded by heat loads used in the original UHS analysis, which contained additional margin. Therefore, the UHS contains sufficient volume to accommodate the MUR-PU heat loads and no system modifications are required to support the MUR-PU.

Therefore, the design and licensing bases of the UHS will continue to be met at MUR-PU conditions.

VI.1.C.iv Residual heat removal

RESPONSE:

The plant cooldown performance licensing basis is documented in Section 5.5.7 of the FSAR. Generally, cooldown times increase due to the higher MUR-PU decay heat load. For the normal (2-train) cooldown cases, the conclusion of cooldown from 350°F to 140°F increases from 34 hours after shutdown (ASD) to 37.7 hours ASD . For the single-train case, conclusion of cooldown from 350°F to 200°F increases from 57.4 hours ASD to 60.1 hours ASD.

One train of equipment can cool the RCS from 350°F to 200°F on RHRS operation within the associated TS limit of 37 hours after shutdown by stopping the operating RCP when RCS temperature reaches 220°F. The calculated cooldown time for this alignment increased from 34.5 hours (current AOR) to 34.6 hours (MUR-PU). The acceptance criterion for single-train cooldown of 37 hours (TS 3.0.3) continues to be met at MUR-PU power uprate conditions.

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

RESPONSE:

The SFP cooling and cleanup system is described in FSAR Section 9.1.3. The SFP is designed to remove decay heat generated by stored spent fuel assemblies from the SFP water. A second function of the system is to maintain clarity (visual) and purity of the SFP water, the transfer canal water, and the refueling water.

The major mechanical components of each unit's system include two SFP pumps, one SFP skimmer pump, one refueling water purification pump, two SFP heat exchangers, one SFP demineralizer, and various filters, skimmers, strainers, and stop valves (both manual and automatic). The system design incorporates two trains of equipment, with either train being capable of removing 100 percent of the design heat load.

Water flows from the SFP to the SFP pump suction, is pumped through the tube side of the SFP heat exchanger, and is returned to the pool. SFP heat exchangers are cooled by component cooling water (CCW). A portion of the SFP water may be diverted through a demineralizer and a filter to maintain SFP water clarity and purity.

There are no changes to the SFP cooling and cleanup system limiting temperatures, pressures, or flow rates as a result of the MUR-PU. System conditions are bounded by the existing system

design conditions. System modifications are not required to support the MUR-PU. The limiting case heat loads at uprate conditions were performed at the higher power level of 2831 MWt, which bounds the MUR-PU power level. There is no change to the loss of cooling analysis. The SFP cooling and cleanup system will be able to maintain adequate water above the fuel to preserve assumptions used in safety analysis of a fuel handling accident. Therefore, the licensing basis for the SFP cooling and cleanup system will be met under MUR-PU conditions.

Refer to Section II Item 31 of this attachment relative to SfP criticality analysis impact due to the MUR-PU.

VI.1.E Radioactive waste systems

RESPONSE:

The Radioactive Waste Systems (Liquid Waste, Gaseous Waste, and Solid Waste) are described in FSAR Chapter 11, Sections 11.2, 11.3, and 11.5, respectively. These systems provide the means to sample, collect, process, store/hold, re-use or release low-level effluents generated during normal operation.

The liquid waste processing system (LWPS) is designed to receive, segregate, process, recycle, and discharge liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are as low as is reasonably achievable.

The LWPS collects and processes potentially radioactive wastes for recycle or for discharge. Provisions are made to sample and analyze fluids before they are recycled or discharged. Based on the laboratory analysis, these wastes are either released under controlled conditions via the cooling water system or retained for further processing. A permanent record of liquid releases is provided by analyses of known volumes of waste.

The existing capacities of various holding, processing, and storage tanks are sufficient at MUR-PU conditions because system flow rates and liquid inventories are not affected by the uprate. The volume of liquid waste primarily depends on reactor coolant bleed, SG blowdown, and leakage from various components. The volume generated during normal operation will not change because of the uprate. Implementing the MUR-PU will not increase the volume inventory of liquid waste processed by the LWPS. The concentration of radioactive nuclides in the LWPS is expected to increase by a maximum of 1.7 percent. This increase in nuclide concentration does not significantly impact LWPS operation.

The GWPS is designed to remove fission product gases from the reactor coolant and has the capacity to contain these throughout the 40-year plant life. This is based on continuous operation with RCS activities associated with operation, with cladding defects in the fuel rods generating 1 percent of the rated core thermal power. The system is also designed to collect and store expected fission gases from the boron recycle evaporator and reactor coolant drain tank throughout the plant life.

The GWPS consists mainly of a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, and gas decay tanks to accumulate the fission product gases. The major input to the GWPS during normal operation is taken from the gas space in the VCT.

The VCT gas space may be purged at a rate of 0.7 scfm. There are no liquid seals in the system.

At MUR-PU conditions, the required containment, confinement, and filtering capacities of the GWPS and the capacities of its various decay and storage tanks are sufficient because the MUR-PU does not materially affect the system flow rates or gas volumes. The MUR-PU may increase radioactivity of gaseous waste a maximum of 1.7 percent. However, system operating procedures can support the potential increase in radioactivity.

Although the GWPS was designed for continuous purge of the VCT and 40-year holdup of fission gases, operating experience at Farley has shown that the GWPS can be operated without a continuous purge while maintaining personnel exposure within limits and maintaining releases within concentration and offsite dose limits.

The solid waste system is designed to transfer spent resins, evaporator concentrates, and chemical tank effluents. This system is installed in Unit 1 and has adequate capacity to serve both units. To provide more efficient solidification and to ensure compliance with current burial ground license requirements (including volume restrictions), provision has been made for the use of a portable cement solidification system. The portable system is operated in the solidification / dewatering facility (SDF) outside the Unit 1 and Unit 2 auxiliary building and is capable of solidifying resins, evaporator concentrates, and chemical drains from both units. The system also serves as a solidification system for the disposable demineralizer system, should solidification be required prior to shipment. A separate system is available to compact dry active wastes such as paper, disposable clothing, rags, towels, floor coverings, shoe covers, plastics, cloth smears, and respirator filters.

Bulk waste may be shipped to a licensed waste processor or to a disposal facility without encapsulation or solidification in accordance with regulations and per applicable license and regulations for the receiver of the waste. During normal work activities, tools, scrap, and other miscellaneous equipment and materials may become radioactively contaminated. The SDF can be used as a decontamination area when needed.

Solidification via the portable system is accomplished with the liner inside a shipping cask or a shielded enclosure in the SDF, which provides the necessary personnel shielding.

The MUR-PU does not have an effect on the generation of solid waste volumes. Therefore, the quantities of low-level, compressible, radioactive wastes (e.g., paper, rags, plastics, clothing, respiratory filters) will not increase because of the uprate. The same is true for high-level wastes such as spent resins and filters. Procedures are in place to segregate, store, classify, package, and track low-level and high-level solid wastes; and because there is no increase in solid waste generation due to uprate, there is not an increase in solid waste storage requirements or shipments from the plant.

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

RESPONSE:

The Farley Units 1 and 2 HVAC systems were evaluated at MUR-PU conditions to determine the effect on the ability of this system to perform its required functions. The following HVAC systems were evaluated:

- Containment Ventilation Systems (Post-Accident Containment Cooling addressed in Section VI.1.B.ii)
- Penetration Room Filtration System
- Containment Preaccess Filtration and Purge Systems
- CR Ventilation Systems
- Auxiliary Building Ventilation Systems
- Radwaste Area
- Turbine Building

The containment ventilation systems are described in FSAR Section 6.2.3. The containment ventilation systems consist of the containment air cooling system, the CRDM cooling system, the reactor cavity cooling system, and the refueling water surface ventilation system. The system is designed to provide containment atmosphere mixing and cooling.

The containment air cooling system consists of four containment air coolers, each with a one-third cooling capacity during normal operation. Up to four units will be operating. Each air cooler consists of a fan and finned tube coil supplied by water from the SWS. During outage periods, auxiliary cooling may be supplied to one of the four containment air coolers to enhance containment heat removal.

The containment air cooling system was analyzed at the MUR-PU conditions and was found to be satisfactory.

The CRDM cooling system provides cooling of the CRDM coils during power operation. The CRDM cooling system consists of fans and ducting to draw air through the CRDM shroud and eject it to the main containment atmosphere. One hundred-percent redundancy is provided by a standby fan.

The CRDM cooling system was analyzed at the MUR-PU conditions and was found to be satisfactory.

The reactor cavity cooling system provides forced convection cooling for the six reactor supports. Design of the reactor cavity cooling system was based on maintaining the localized concrete temperature at less than 200°F. The RV support cooling system, consisting of two 100 percent capacity fans and ducting, is arranged to cool the RV supports by drawing air through the supports. One hundred percent redundancy of all active components is provided.

Cooling is supplied to the reactor cavity by the discharge from the containment cooling system ductwork at the 155-foot elevation. The reactor supports are positioned underneath the RV hot

and cold leg nozzles. Air is drawn upward from the bottom of the reactor cavity past the vessel (along the outside of the downcomer and lower plenum) and out the inspection openings above the nozzles. Since these areas of the vessel are at T_{cold} and T_{cold} is decreasing for the MUR-PU, the cavity fans will continue to provide adequate cooling for MUR-PU.

During refueling operations, the function of the refueling water surface ventilation system is to remove water vapor above the refueling cavity pool to improve core visibility, reduce personnel heat stress, and reduce personnel inhalation exposure. A supply fan draws air from the containment atmosphere and delivers it above the water surface from one side of the canal. An exhaust fan draws the air from the opposite side of the canal and exhausts to the purge system where it is diluted, filtered, and discharged from the plant vent.

The source terms for the MUR-PU are bounded by the current analysis. Therefore, the capability of the ventilation system to protect personnel will not be challenged. With respect to the additional decay heat, RCS temperature during refueling is limited by TSs such that the heat removal capability of the ventilation system will not be challenged and there will be no additional water vapor.

The penetration room filtration system is described in FSAR Section 6.2.3. The primary purpose of the penetration room filtration system is to limit release to the environment of radioisotopes which may leak from containment into the penetration room under accident conditions including a LOCA. Although not credited for the fuel handling accident dose analysis, the penetration room filtration system can be aligned to process air from the fuel handling area in the event of a fuel handling accident. The system is designed to conform to RG 1.52.

The post-accident temperature and pressure profile remained conservative after MUR-PU. Also, there is no increase in amount of potential radioisotopes that could leak from containment. Therefore, the penetration room filtration system is able to perform its design function at MUR-PU.

The containment pre-access filtration and purge systems are described in FSAR Section 6.2.3. The containment pre-access filtration system is designed to reduce the airborne fission product activity in the containment atmosphere prior to personnel access at normal power conditions or before a plant outage. With the use of the filtration system, management of the radiological dose and outage time can be better controlled and limited. Use of the containment pre-access filtration system is not required for containment occupancy as the containment mini-purge system is designed to maintain radioactivity levels inside containment consistent with occupancy requirements. Use of the pre-access filtration system would provide for reduced usage of the mini-purge system.

The function of the purge/mini-purge systems is to prevent pressure build-up in containment and limit build-up of noble gases, iodine, and particulates to provide an acceptable working environment inside containment. The pre-access system and purge systems in combination, provide sufficient circulation/filtering throughout containment to allow safe access to containment.

The source terms for the MUR-PU are bounded by the current analysis. The ability of the preaccess filtration system and mini-purge system to meet the criteria is not affected by the MUR-PU. For the containment pre-access filtration, the MUR-PU is not changing the limiting containment conditions assumed for normal operation, current capacity is adequate for the

MUR-PU. The pre-access filter system is not required for accident mitigation or for long-term post-accident containment mixing/venting. The MUR-PU will not result in more limiting atmospheric conditions in any mode of operation such that the pressure reducing capability of the purge/mini purge systems will not be challenged.

The CR air ventilation systems consisting of the CR air conditioning and filtration systems are described in FSAR Section 9.4. The CR air conditioning and filtration systems are designed with sufficient redundancy and separation of components to provide reliable operation under normal conditions and guarantee operation under emergency conditions. The two 100 percent capacity, Category I AC package units are each designed to maintain CR temperature at approximately 78°F. The safety related components in the CR are designed to withstand a maximum environmental temperature of 120°F.

The MUR-PU will not affect normal ambient conditions outside containment, the MSLB M&E releases calculated for MUR-PU are bounded by current analyses, and radiological consequences for postulated accidents are well within design limits. Therefore, the CR HVAC system will continue to perform its design function under normal/accident conditions. The CR air conditioning and filtration system is not impacted by the MUR-PU.

The auxiliary building ventilation system is described in FSAR Section 9.4 and is designed to provide a suitable environment for equipment and personnel. The system provides maximum safety and convenience for operating personnel by arranging the ventilation equipment in zones so that potentially contaminated areas are separated from clean areas. The path of ventilating air is from areas of low activity toward areas of progressively higher activity. Separate heating and ventilating systems serve the radioactive areas and nonradioactive areas, including the lower equipment rooms, technical support center, and fuel handling areas of the auxiliary building. The computer room, access CR, electrical equipment rooms, and technical support center have individual air conditioning systems.

The non-radioactive area will not see increased heat loads due to the MUR-PU. As such, the current ventilation system is adequate. The non-radioactive area ventilation system is not an engineered safety feature and no credit is taken for its operation in analyzing the consequences of any accident. The nonradioactive area ventilation system is not impacted by the MUR-PU.

The fuel handling area heating, ventilating and filtration system is not used to reduce accident doses and the isolation function is not impacted by the MUR-PU. As such, the only design requirement potentially impacted by the MUR-PU is the maximum ambient temperature due to possible increased decay heat levels in the SFP. The SFP cooling analysis (Section 11.7, "Cooling Water Systems") concluded there is negligible change due to the MUR-PU. Therefore, no design limits will be challenged for the MUR-PU. With respect to habitability, decay heat levels will increase only slightly and should not represent a burden to personnel.

The computer room HVAC system maintains the environment between 60° and 80°F and 50 percent relative humidity which guarantee both the comfort and safety of the operators and the integrity of the computer room components. The system also provides sufficient air capacity to maintain the computer room at a slightly positive pressure. The MUR-PU is unrelated to the cooling requirements of the computer room or the ability of the system to maintain positive pressure. This system is not an engineered safety feature and no credit is taken for its operation

in analyzing the consequences of any accident. The computer room ventilation system is not impacted by the MUR-PU.

The access control HVAC system is not an engineered safety feature and no credit is taken for its operation in analyzing the consequences of any accident. Therefore, there are no additional challenges to this system for the MUR-PU and the current ventilation system will continue to perform its design function.

A separate and independent AC system is used in the electrical cable spreading room and the 600-V load center rooms. There are no changes to the normal or post-accident heat loads in these rooms for the MUR-PU. The normal ventilation system is not an engineered safety feature and no credit is taken for its operation in analyzing the consequences of any accident. Therefore, there are no additional challenges to this system for the MUR-PU and the current ventilation system will continue to perform its design function.

The battery room exhaust system draws air from each battery room by an independent exhaust fan through Seismic Category I exhaust ducts during normal and accident modes of battery operation. Power supplies to the fans are Class 1 E electric systems. Makeup air to each room is directly supplied from the non-radioactive air handling unit. Since there are no changes to the DC system, the uprate will not change normal ambient conditions, and the M&E releases calculated for uprate are bounded by current analyses, this system will not be further challenged due to the MUR-PU.

For the Battery Charger Room, Motor Control Center, and 600-V Load Center Room, floor- or ceiling-mounted coolers with horizontal or vertical air handling units are provided. Cooling water is supplied by SW. A temperature switch located in each room energizes the respective cooler when the setpoint is reached. The coolers are designed to maintain ambient temperature to no more than 104°F for equipment operability. Note that in the evaluation supporting the 106.2°F post-accident SW temperature for these rooms, calculated room temperatures exceeded 104°F. However, per FSAR Table 3. I I-1, Note J, the equipment in these rooms is evaluated when temperatures exceed 104°F. Further, the calculated room temperatures are within the limits of the TRM (Unit 2 only) for both normal and accident conditions. Since maximum SW temperature is not increasing and the heat loads in these rooms are not increasing, the cooling systems will not be further challenged due to the MUR-PU.

The Sampling Room, Gas Analysis Room, Counting Room, and Radioactive Laboratory Heating and Air Conditioning System is designed to maintain the sampling room at 80°F and the remaining rooms at 75°F. These systems are not engineered safety features and no credit is taken for their operation in analyzing the consequences of any accident. The power uprate is unrelated to the cooling requirement of these areas. Therefore, there are no additional challenges to this system for uprate and the current ventilation system will continue to perform its design function.

The Engineered Safety Feature Pump Room Coolers are designed to maintain the ambient temperature in each of the charging/high head, RHR, CS, CCW, and AFW pump rooms at or below 104°F for normal operation. The current SW temperatures remain bounding for the MUR-PU. Therefore, no additional challenges to this system will result due to the MUR-PU.

The Technical Support Center HVAC System is comprised of a safety-related power supply bus, an air handling unit, a charcoal filter, and related ductwork. During normal operation, outside air

is drawn in, mixed with recirculated air, then cooled or heated to maintain design temperature (75°F and 50 percent relative humidity). When high radiation in detected in the CR, the system automatically diverts the fresh/recirculated air mixture through the charcoal filter then into the air handling unit just prior to entering the technical support center. A positive pressure is maintained in the support center to prevent inleakage. The HVAC system is designed to allow access during accident conditions. However, although the power supply meets safety-related design criteria, the HVAC system itself is not classified nor is it redundant. Since the uprate will not change normal ambient conditions and the M&E releases calculated for the uprate are bounded by current analyses, this system will not be further challenged due to the MUR-PU.

The Radwaste Area heating, ventilating and filtration system is described in FSAR Section 9.4.3. The Radwaste Area system is an independent system designed to control and direct all potentially contaminated air to the vent stack via pre-filter, HEPA, and charcoal filters. The supply air handling unit provides once through filtered and tempered outside air (or a mix of outside and recirculated air) to all personnel occupancy areas, the monitor tank compartments, and treated waste holdup tank areas. The system is designed to provide a suitable environment for equipment and personnel by limiting maximum ambient temperature to 110°F when the outdoor temperature is 95°F, maintain minimum temperature to 65°F when the outdoor temperature is 20°F, and maintain a slightly negative pressure with respect to the surrounding corridors to prevent outleakage of contaminants. Any variation in temperatures due to uprate of the liquid processed through equipment in the associated rooms will be insignificant with respect to the heat load to the rooms. This system is not safety-related and no credit is taken for its operation in mitigating the consequences of any accident. Therefore, the MUR-PU will not challenge the capacity of this system under either normal or accident conditions.

The TB Heating and Cooling system is described in FSAR Section 9.4.4. The primary sources of heat load in the TB are controlled by conditioning the air using the closed loop chilled water system (CWS). This system was evaluated for increases resulting from the MUR-PU. It was determined that the uprate conditions will not significantly increase the heat load to the TB. Localized hot spots at sensitive components should experience slightly higher temperatures during operation at uprated conditions without shortening the expected life of these components. The TB CWS will not be further challenged due to uprate.

References for Section VI:

None

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-PU will be implemented under the administrative controls of the SNC design change process. The design change process ensures any impacted normal operating procedures, abnormal operating procedures (AOPs), and emergency operating procedures (EOPs) having operator actions are revised prior to the implementation of the MUR-PU if required. An evaluation was performed of the Operator Actions and no impacts were identified.

Time Critical Operator Actions (TCOAs) are associated with the mitigation of postulated events. These actions must be performed in a specified time in order to assure the plant complies with assumptions made during the analysis of these postulated events. The TCOAs were evaluated individually in system evaluations and against the Farley licensing analyses presented in Section II 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 EOPs and AOPs have been reviewed for power uprate impacts, and no impacted procedures have been identified. Additionally, the uprate is being implemented under the administrative controls of the design change process. The design change process ensures that any impacted procedures will be revised prior to the implementation of the power uprate.

VII.2.B Control room controls, displays (including the safety parameter display system) and alarms

RESPONSE:

The physical modifications to the plant required to support the Farley MUR-PU include the installation of the Cameron LEFM Check-Plus feedwater ultrasonic measurement system and the new high pressure turbine. Additionally, a review of plant systems has indicated that only minor modifications are necessary (e.g., software modification that redefines the new 100-percent RTP). Farley follows the established design change process and procedures (as noted in Section VII.2.A) to ensure that the necessary minor modifications are installed prior to implementing the proposed power uprate. An "LEFM System Trouble" alarm window will be

added to the CR alarm panel to alert the operator when there is a problem with the LEFM. The new high pressure turbine modifications require no changes to the CR controls, displays, or alarms.

VII.2.C Control room plant reference simulator

RESPONSE:

As part of the Farley design change process and procedures, a review of the plant simulator is conducted, and necessary changes resulting from the MUR-PU (including changes for the new calorimetric and displays for the LEFM interface) to the Farley simulator are identified. The design change process ensures that the simulator modifications are made prior to the implementation of the uprate.

VII.2.D Operator training program

RESPONSE:

The Operations Training department has been involved in the design review process for the modifications required to support the MUR-PU. The Operations staff are trained on the modifications, TS changes, and procedure changes prior to implementation of the MUR-PU. Training on the operation of the Cameron LEFM Check-Plus system and calorimetric impacts are also developed and completed prior to implementation of the MUR-PU.

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 as discussed above (including changes to the simulator and the associated manuals and instructional materials) are implemented in accordance with the Farley engineering design change process (as noted in Section VII.2.A). SNC will complete all modifications identified in Section VII.2.B related to the MUR-PU and complete the training of operators prior to implementation of the power uprate. Plant modifications are evaluated to ensure that changes in Operator actions do not adversely affect defense in depth or safety margins.

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:

The unit operating procedure for Mode 1 includes precautions for temporary operation above the licensed power level for certain periods of time. These precautions will be revised to account for the MUR-PU power level.

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.

VII.5.A.i <u>Non-Radiological Effluents</u>

RESPONSE:

10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment if facility operation per the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

The proposed change does not involve installing new equipment or modifying any existing equipment that might affect the types or amounts of effluents released offsite.

There will be no significant change in the types or significant increase in the amounts of non-radiological effluents released offsite during normal operation.

Chemical and thermal discharges, including any minor impacts that may arise from the MUR-PU, are administratively controlled by plant procedures and subject to the existing requirements of the Farley NPDES permit.

VII.5.A.ii **Radiological Effluents:**

RESPONSE:

Operation at the MUR-PU would result in a slight increase in long-lived effluent isotopic releases and doses, approximately proportional to the MUR-PU percentage increase.

It is noted that the Emergency Action Levels (EALs) are not dependent on reactor power levels and therefore are not impacted by the MUR-PU.

The radioactive gaseous and liquid waste systems are adequately designed to process the increased radioactivity released due to operation at MUR-PU. The liquid and gaseous radwaste effluent treatment systems, in conjunction with the procedures and controls provided by the ODCM, will remain capable of maintaining normal operation offsite doses well within the regulatory requirements of 10 CFR 20, 10 CFR 50 Appendix I and 40 CFR 190.

The design of the Farley ISFSIs will be capable, in conjunction with the procedures and controls provided by the ODCM, of maintaining both onsite and offsite doses within the regulatory requirements of 10 CFR 72.104.

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

RESPONSE:

Normal Operation Radiation Levels

The increase in reactor power from the current licensed level of 2775 MWt to the MUR-PU core power level of 2821 MWt will result in a corresponding 1.7-percent increase in the neutron and gamma flux in and around the core, fission product, and actinide activity inventory in the core and spent fuels; N-16 source in the reactor coolant; neutron activation source in the vicinity of the reactor core; and fission/corrosion product activity in the reactor coolant and downstream systems.

For the same source-shield-detector configuration, the dose rate at a given detector point is directly proportional to the radiation source strength in the source region. Because there is no change in the fuel cycle length, the normal operation radiation levels in most of the plant areas are expected to increase by approximately 1.7 percent (i.e., the percentage of the core power uprate). The exposure to plant personnel and to the offsite public due to direct shine is also expected to increase by the same percentage. As discussed in the radiation zone and shielding adequacy review below, the MUR-PU will not require additional radiation shielding to support normal plant operation, and existing plant design and the Radiation Protection Program will ensure that operator and offsite public exposure is maintained within the requirements of 10 CFR 20.

Radiation Zoning and Shielding Adequacy

Shielding is used to reduce radiation dose rates in various parts of the station to acceptable levels that are consistent with operational and maintenance requirements and also below the limits allowed for continuous non-occupational exposure at the site boundary.

The original Farley shielding design was based on generalized occupancy requirements in various radiation zones of the station and on conservative radiation source terms at nominal full power conditions and a design basis normal operation reactor coolant source term based on one-percent fuel defects.

The plant occupancy requirements are not affected by the MUR-PU. The layout/configuration of systems containing radioactivity are also not affected by the MUR-PU. Plant operation at the MUR-PU core power level of 2821 MWt represents an increase in power level of about 1.7 percent. The increase in expected radiation levels due to the MUR-PU will not affect radiation zoning or shielding requirements in the various areas of the plant. This is because the increase is offset by the following:

- Conservative analytical techniques typically used to establish shielding requirements, such as ignoring the shadow shielding effect of neighboring sources, rounding up the calculated shield thickness to a higher whole number, using conservative buildup factors, and similar techniques.
- Conservatism in the original design basis reactor coolant source terms used to establish the radiation zones (assumed one-percent fuel defects).
- FNP TS 3.4.16, "Reactor Coolant System – Specific Activity" which limits the reactor coolant concentrations to levels well below the original design basis source terms.

Additionally, individual worker exposures are maintained within regulatory limits and ALARA by the site Radiation Protection Program, which controls operator exposure by controlling access to radiation areas and maintains compliance with 10 CFR 20.

The MUR-PU will not require additional radiation shielding to support normal plant operation, and existing plant design and the Radiation Protection Program will ensure that operator exposure is maintained within the requirements of 10 CFR 20.

VII.6 Programs and Generic Issues

VII.6.A Fire Protection Program

RESPONSE:

The Farley Fire Protection Program (FPP) is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) property loss prevention standards and related industry standards. With regard to NRC criteria, the FPP meets the requirements of 10 CFR 50.48(c), which endorses (with exceptions) NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. Farley Units 1 and 2 have further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," as endorsed by NRC RG 1.205, "Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants."

The station FPP consists of activities and functions that are performed to minimize the probability and consequences of a postulated fire. In the event of a fire, the program and system

designs ensure the capability to shut down the reactor and maintain it in a safe and stable (achieve and maintain safe shutdown) shutdown condition. This includes the following:

- Classical fire protection elements such as fire detection and suppression systems and equipment (including active as well as passive design features) and programmatic / organizational elements for minimizing the chance of fire occurring and ensuring minimal impact should one occur
- Nuclear safety compliance assessments, safe shutdown capability, and Farley's design capability to achieve nuclear safety capability safe and stable conditions in the event of a single damaging fire
- A supporting fire probabilistic risk assessment that provides both overall and detailed risk insights
- Radioactive release
- Non-power operations

The FPP is based on the NRC's requirements and guidelines, NFPA 805, NEIL property loss prevention standards, and related industry standards. The FPP design basis contains various references to the calculations that demonstrate compliance to these requirements. These calculations were reviewed for any impact due to the MUR-PU (i.e., change in temperature, pressure, flow, or reactor power level). The reviewed calculations do not consider reactor power level or the operating conditions of supporting systems. The change in reactor power level and operating conditions of systems do not affect FPP compliance with required regulations and guidelines. Therefore, the FPP is not impacted by the MUR-PU.

There are no FPP recommendations, and there are no recommended plant modifications to the system required to support the MUR-PU. Any changes to combustible loadings resulting from the installation of equipment and components installed in support of the MUR-PU will be evaluated and controlled under the Farley design change process.

VII.6.B Containment Coatings Program

RESPONSE:

The containment coatings program is administered by SNC Corporate Procedure NMP-MA-011 and its sub-tier documents. The program was developed to meet the criteria of ANSI Standard N101.2. Whereas plant coatings programs are typically governed by RG 1.54 and related ANSI Standard N101.4, Farley is not subject to the requirements of these documents because the requirements postdate the Farley construction permits. However, the qualification testing of Service Level I coatings used for new applications or repair/replacement inside the Farley Units 1 and 2 containment buildings meets the intent of ANSI N101.2. This is consistent with the Farley Units 1 and 2 response to NRC GL 98-04, in which SNC stated that Farley Units 1 and 2 are not subject to either RG 1.54 or related ANSI Standard N101.4 but rather meet the criteria given in the ANSI Proposed Standard N101.2-1971.

The Farley Units 1 and 2 response to NRC GL 98-04 also states that Farley Units 1 and 2 meet the requirements of 10 CFR 50.46(b)(5) for long-term cooling. The containment coatings program documents and the FSAR were reviewed and found to be unaffected by the MUR-PU.

Therefore, the containment coatings program for Service Level I coatings inside containment will continue to comply with the licensing bases of NRC GL 98-04 and will continue to meet the intent of ANSI N101.2. In addition, the program will continue to meet the requirements of 10 CFR 50.46(b)(5).

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

RESPONSE:

The NRC issued GL 89-10 requiring licensees to develop a comprehensive program to ensure motor-operated valves (MOVs) in safety-related systems would operate under design basis conditions. The engineering review determined that the systems' worst-case operating conditions are not impacted by the MUR-PU. Therefore, the maximum differential pressures expected across the valves in the MOV program are not affected. The values for these parameters at current conditions bound the values at MUR-PU conditions. Therefore, these parameters do not affect the calculations that determine MOV thrust and torque values. System flow rates are also not significantly impacted by the MUR-PU. Therefore, the MOV flow rates documented in the design basis review calculations for the GL 89-10-identified MOVs at current conditions will continue to bound the flow rates at MUR-PU conditions. The MUR-PU does not affect the maximum ambient temperatures used to determine MOV motor capability torque values at current conditions. Therefore, the MOV program will continue to comply with the licensing basis requirements of GL 89-10 after implementation of the MUR-PU.

The NRC issued GL 95-07 to address potential pressure locking and thermal binding of safety-related power-operated gate valves. The engineering review determined that there are no significant changes in system operating parameters, fluid differential pressures at the valves, or ambient temperatures. Therefore, the MUR-PU does not affect the pressure locking evaluations previously completed. The thrust required to open the applicable valves will remain less than the motor actuator capabilities at MUR-PU conditions. The MUR-PU does not affect valve design, valve function, or operational conditions. New conditions were not created that would affect valve susceptibility to pressure locking or thermal binding. Therefore, valve pressure locking and thermal binding susceptibility are not impacted by the MUR-PU and the MOV program will continue to comply with the licensing basis requirements of GL 95-07.

The NRC issued GL 96-05 to require licensees to develop programs for periodic verification of design basis capability of safety-related MOVs. No MOVs are required to be added to the MOV program as a result of the MUR-PU. The engineering review determined that the systems' worst-case operating conditions are not impacted by the MUR-PU. Therefore, the maximum differential pressures expected across the valves are not affected. The values for these parameters at current conditions bound the values at MUR-PU conditions. Any changes in the periodic verification requirements as a result of changes in risk category due to the MUR-PU will be addressed in accordance with the SNC MOV program requirements. Therefore, the MOV program will continue to comply with the licensing basis requirements of GL 96-05 after implementation of the MUR-PU.

The SNC air-operated valve (AOV) program includes the following categories of AOVs:

1. Category 1: AOVs that are safety-related, active, and high safety-significance; or AOVs that are non-safety related, active, and high safety-significance.

2. Category 2: AOVs that are safety-related and active, but do not have high safety significance.
3. Category 3: AOVs that are not Category 1 or 2 and have been determined by the AOV Expert Panel to be Category 3. Categorization of Category 3 valves considers available maintenance rule scope, probability risk assessment ranking, and equipment reliability classifications information. Categorization into Category 3 also considers valves that could cause a plant trip or load reduction or affect thermal performance if they do not perform properly.
4. Category 4: AOVs that are not Category 1, 2, or 3. AOVs in this category may or may not be safety-related. They have been determined to be of lesser safety significance, and they do not have a significant potential for affecting plant operation.

An evaluation was conducted to determine the impact on AOV performance due to the MUR-PU. The MUR-PU does not affect the systems' worst-case design conditions. Therefore, the calculations that determine the required AOV thrust and torque values are not affected. There is no impact on the compressed air system. The capability of the AOV actuators to produce the required thrust/torque is not affected. There are no changes to the AOV risk categories. The analytical methodology, testing methodology, and testing frequencies are not affected. No changes are required to the existing AOV program. Therefore, the existing AOV program remains valid at MUR-PU conditions.

VII.6.D Containment Leakage Rate Testing Program

RESPONSE:

The Appendix J (ILRT/LLRT) Containment Leakage Rate Testing Program is implemented in accordance with 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. The Farley program documents, FSAR, and TSs all conform to the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, and are unaffected by the MUR-PU. Therefore, the Appendix J program will continue to comply with the licensing bases of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J.

References for Section VII:

None

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 TS changes is provided in Section 2 of this Enclosure, consistent with SNC License Amendment Request format. Amended TSs are attached, with a marked-up copy in Attachment 1.

VIII.1.B Identification of analyses affected by and/or supporting the change

RESPONSE:

The heat balance uncertainty has been revised to reflect 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 Attachment.

VIII.1.C Justification for the change, including the type of information discussed in Section III, for any analyses that support and/or are affected by change.

RESPONSE:

The justification for the TS changes is provided in Section 3 of this Enclosure, consistent with SNC License Amendment Request format.

References for Section VIII:

None

Attachment 5 contains proprietary and non-proprietary versions of report ER-1180.
Under 10 CFR 2.390, withhold from public disclosure ER-1180P Rev. 1.
Upon removal of ER-1180P Rev. 1, Attachment 5 is uncontrolled.

Attachment 5

Engineering Report: ER-1180P/NP Rev. 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM✓+ System" & Caldon Application for Withholding Proprietary Information from Public Disclosure CAW-19-01, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice

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CAMERON
A Schlumberger Company

August 21, 2019
CAW 19-01

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1180 Rev 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM ✔ + System"

Gentlemen:

This application for withholding is submitted by Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Technologies US, Inc., 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 19-01 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 19-01 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

Schlumberger-Private

August 21, 2019
CAW 19-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Joanna Phillips, who, being by me duly sworn according to law, deposes and says that she is authorized to execute this Affidavit on behalf of Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Technologies US, Inc., and that the averments of fact set forth in this Affidavit are true and correct to the best of her knowledge, information, and belief:

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 21st day of

August, 2019

Frances A. Lewis
Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

Schlumberger-Private

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

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

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a

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system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

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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.

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(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Cameron Engineering Report ER-1180 Rev 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM ✔ + System"

- Table of Contents page contains partial proprietary information
- Pages 4, 5, 7, and 8 contain partial proprietary information
- Appendix A, A.4 and A.5 cover pages contain partial proprietary information
- Appendices A.1, A.2, A.4, A.5 and B are proprietary in their entirety

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Farley Unit 1 for flow measurement at the licensed reactor thermal power level of 2821 MWt.

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

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

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

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Further the deponent sayeth not.

Schlumberger-Private

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CAMERON
A Schlumberger Company

August 21, 2019
CAW 19-02

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1181 Rev 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM ✓ + System"

Gentlemen:

This application for withholding is submitted by Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Technologies US, Inc., 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 19-02 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 19-02 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

Schlumberger-Private

August 21, 2019
CAW 19-02

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Joanna Phillips, who, being by me duly sworn according to law, deposes and says that she is authorized to execute this Affidavit on behalf of Cameron (Holding) Corporation, a Nevada Corporation (herein called "Cameron") on behalf of its operating unit, Caldon Technologies US, Inc., and that the averments of fact set forth in this Affidavit are true and correct to the best of her knowledge, information, and belief:

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 21st day of

August, 2019

Frances A. Lewis
Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

Schlumberger-Private

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

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

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a

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system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

Schlumberger-Private

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Cameron Engineering Report ER-1181P Rev 1 "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM ✔ + System"

- Table of Contents page contains partial proprietary information
- Pages 4, 5, 7, and 8 contain partial proprietary information
- Appendix A, A.4 and A.5 cover pages contain partial proprietary information
- Appendices A.1, A.2, A.4, A.5 and B are proprietary in their entirety

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Farley Unit 2 for flow measurement at the licensed reactor thermal power level of 2821 MWt.

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

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

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

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Further the deponent sayeth not.

Schlumberger-Private

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CAMERON
A Schlumberger Company

August 21, 2019
CAW 19-03

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1182 Rev 1 "Meter Factor Calculation and Accuracy Assessment for Farley Unit 1"

Gentlemen:

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

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 19-03 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 19-03 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

Schlumberger-Private

August 21, 2019
CAW 19-03

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 21st day of

August, 2019

Frances A. Lewis

Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

Schlumberger-Private

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

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

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

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

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a

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system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

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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.

Schlumberger-Private

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Cameron Engineering Report ER-1182P Rev 1 "Meter Factor Calculation and Accuracy Assessment for Farley Unit 1"

- Pages 1, 2, 3, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 20, 21, 22, 23, 24 contain partial proprietary information
- Appendix B index page and Appendix A, B.3 and B.4 cover pages contain partial proprietary information
- Appendices A, B.1, B.2, B.3, and B.4 are proprietary in their entirety

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Farley Unit 1 for flow measurement at the licensed reactor thermal power level of 2821 MWt.

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

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

In order for competitors of Cameron to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant

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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.

Schlumberger-Private

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CAMERON
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August 21, 2019
CAW 19-04

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Cameron Engineering Report ER-1183 Rev 1 "Meter Factor Calculation and Accuracy Assessment for Farley Unit 2"

Gentlemen:

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

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 19-04 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 19-04 and should be addressed to the undersigned.

Very truly yours,

Joanna Phillips
Nuclear Sales Manager

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

Schlumberger-Private

August 21, 2019
CAW 19-04

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Joanna Phillips
Nuclear Sales Manager

Signed and sworn to before me

this 21st day of

August, 2019

Frances A. Lewis
Notary Public

Commonwealth of Pennsylvania - Notary Seal
Frances A. Lewis, Notary Public
Allegheny County
My commission expires November 25, 2022
Commission number 1287160
Member, Pennsylvania Association of Notaries

Schlumberger-Private

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

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

3. I have personal knowledge of the criteria and procedures utilized by Cameron in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Cameron requests that the information identified in paragraph 5(v) below be withheld from the public on the following bases:

   Trade secrets and commercial information obtained from a person and privileged or confidential

   The material and information provided herewith is so designated by Cameron, in accordance with those criteria and procedures, for the reasons set forth below.

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a

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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.

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(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (a), (b) and (c), above.

Schlumberger-Private

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(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.

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(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.

Schlumberger-Private

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is the submittal titled:

Cameron Engineering Report ER-1183P Rev 1 "Meter Factor Calculation and Accuracy Assessment for Farley Unit 2"

- Pages 1, 2, 3, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 20, 21, 22, 23, 24 contain partial proprietary information
- Appendix B index page and Appendix A, B.3 and B.4 cover pages contain partial proprietary information
- Appendices A, B.1, B.2, B.3, and B.4 are proprietary in their entirety

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus System used by Farley Unit 1 for flow measurement at the licensed reactor thermal power level of 2821 MWt.

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

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manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

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