J. Todd Conner
Site Vice President

DTE Energy Company
6400 N. Dixie Highway, Newport, MI 48166
Tel: 734.586.4849 Fax: 734.586.5295
Email: connerj@dteenergy.com

DTE Energy

PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390

10 CFR 50.90

February 7, 2013
NRC-13-0004

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

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

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

Subject: License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

In accordance with the provisions of Section 50.90 and Appendix K to Part 50 of Title 10, *Code of Federal Regulations* (10 CFR), DTE Electric Company (DTE) is submitting a request for an amendment to the Operating License and Technical Specifications (TS) for Fermi 2.

The proposed amendment would revise the Fermi 2 Operating License and TS to implement an increase of approximately 1.64% in rated thermal power from the current licensed thermal power (CLTP) of 3430 megawatts thermal (MWt) to 3486 MWt. The proposed changes are based on increased feedwater flow measurement accuracy, which was achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The LEFM instrumentation was installed at Fermi 2 in 2010. The content of this request is in accordance with the guidance contained in Reference 2.

Enclosures 7, 10 and 11 contain Proprietary Information – Withhold Under 10 CFR 2.390. When separated from the enclosures, this document is decontrolled.

performance during testing that verifies instrument channel setting values established by the plant specific setpoint methodologies.

- Enclosure 1 provides a description and evaluation of the proposed changes.
- Enclosure 2 provides the marked up pages of existing Operating License and TS to show the proposed changes.
- Enclosure 3 provides a markup of the existing Technical Requirements Manual (TRM) and TS Bases to show the proposed changes. These pages are provided for information only and do not require NRC approval.
- Enclosure 4 provides the revised (clean) Operating License and TS pages.
- Enclosure 5 provides a cross reference between the contents of this request and RIS 2002-03 (Reference 2).
- Enclosure 6 provides a summary of the regulatory commitments made in this request.
- Enclosure 7 provides the General Electric-Hitachi (GEH) Nuclear Energy document NEDC-33578P, "Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization," (Proprietary Version).
- Enclosure 8 provides affidavits from GEH Nuclear Energy and the Electric Power Research Institute (EPRI) supporting the withholding of information in Enclosure 7 from public disclosure.
- Enclosure 9 provides the General Electric-Hitachi (GEH) Nuclear Energy document NEDO-33578, "Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization," (Non-Proprietary Version).
- Enclosure 10 provides Cameron document ER-781, Revision 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Fermi Unit 2 Nuclear Generating Station Using the LEFM √ + System," (Proprietary Version).
- Enclosure 11 provides Cameron document ER-818, Revision 0, "Meter Factor Calculation and Accuracy Assessment for Fermi Unit 2," (Proprietary Version).
- Enclosure 12 provides affidavits from Cameron International Corporation supporting withholding of information in Enclosures 10 and 11 from public disclosure.
- Enclosure 13 provides Fermi 2 calculation DC-6443, Volume I DCD 1, Revision A, "Reactor Core Thermal Power Uncertainty with Feedwater Flow Measured by LEFM CheckPlus C System."
- Enclosure 14 provides Fermi 2 calculation DC-4608, Volume I DCD 1, Revision 0, "NUMAC Power Range Neutron Monitoring System (PRNM) Surveillance Validation," addressing the Average Power Range Monitors Simulated Thermal Power - Upscale trip setpoint revisions.
- Enclosure 15 provides drawings describing the installation of the LEFM.

The proposed changes have been reviewed by the Fermi 2 Plant Onsite Review Organization and the Fermi 2 Nuclear Safety Review Group in accordance with the requirements of the Fermi 2 Quality Assurance Program.

The proposed changes have been reviewed by the Fermi 2 Plant Onsite Review Organization and the Fermi 2 Nuclear Safety Review Group in accordance with the requirements of the Fermi 2 Quality Assurance Program.

DTE requests approval of the proposed license amendment by November 7, 2013, with the amendment being implemented upon startup from the Sixteenth Refueling Outage. The Sixteenth Refueling Outage is currently scheduled to begin on February 10, 2014.

In accordance with 10 CFR 50.91(a)(1), "Notice for Public Comment," the analysis about the issue of no significant hazards consideration using the standards in 10 CFR 50.92 is being provided to the Commission in accordance with the distribution requirements in 10 CFR 50.4.

In accordance with 10 CFR 50.91(b)(1), "State Consultation," a copy of this application and its reasoned analysis about no significant hazards considerations is being provided to the designated Michigan state official.

In accordance with 10 CFR 2.390, DTE requests withholding of Enclosures 7, 10 and 11 from public disclosure. Enclosure 7 contains information that is considered proprietary by GEH Nuclear Energy and the Electric Power Research Institute (EPRI). Affidavits supporting this request are provided in Enclosure 8 and a non-proprietary version of Enclosure 7 is provided in Enclosure 9. Enclosures 10 and 11 are considered proprietary by Cameron International Corporation. Affidavits supporting these requests are included in Enclosure 12. Non-proprietary versions of Enclosures 10 and 11 are not available.

Should you have any questions or require additional information, please contact Mr. Zachary W. Rad of my staff at (734) 586-5076.

Sincerely,

Enclosures:

1. Evaluation of the Proposed License Amendment
2. Markup of Existing Operating License and Technical Specifications
3. Markup of Existing Technical Requirements Manual and Technical Specifications Bases
4. Revised (Clean) Operating License and Technical Specifications Pages
5. RIS 2002-03 Cross Reference
6. Summary of Regulatory Commitments
7. General Electric-Hitachi (GEH) Nuclear Energy Document NEDC-33578P, "Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization," (Proprietary Version)
8. Affidavits from GEH Nuclear Energy and EPRI
9. General Electric-Hitachi (GEH) Nuclear Energy Document NEDO-33578, "Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization," (Non-Proprietary Version)
10. Cameron Document ER-781, Revision 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Fermi Unit 2 Nuclear Generating Station Using the LEFM √ + System," (Proprietary Version)
11. Cameron Document ER-818, Revision 0, "Meter Factor Calculation and Accuracy Assessment for Fermi Unit 2," (Proprietary Version)
12. Affidavits from Cameron International Corporation
13. Fermi 2 Calculation DC-6443, Volume I DCD 1, Revision A, "Reactor Core Thermal Power Uncertainty with Feedwater Flow Measured by LEFM CheckPlus C System"
14. Fermi 2 Calculation DC-4608, Volume I DCD 1, Revision 0, "NUMAC Power Range Neutron Monitoring System (PRNM) Surveillance Validation"
15. Drawings Describing the Installation of the LEFM

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, J. Todd Conner, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

J. Todd Conner
Site Vice President
Nuclear Generation

On this 7 day of February, 2013 before me personally appeared J. Todd Conner, being first duly sworn and says that he executed the foregoing as his free act and deed.

Sharon S. Marshall
Notary Public

SHARON S. MARSHALL
NOTARY PUBLIC, STATE OF MI
COUNTY OF MONROE
MY COMMISSION EXPIRES Jun 14, 2013
ACTING IN COUNTY OF Monroe

Enclosure 1
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Evaluation of the Proposed License Amendment

28 Pages

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1.0 Description

In accordance with 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit" and 10 CFR 50, Appendix K, "ECCS Evaluation Models," DTE Electric Company (DTE) requests an amendment to Facility Operating License No. NPF-43 for Fermi 2. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of approximately 1.64% in Rated Thermal Power (RTP) from 3430 megawatts thermal (MWt) to 3486 MWt.

The proposed changes are based on increased feedwater flow measurement accuracy, which is achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation. The LEFM instrumentation was installed at Fermi 2 in 2010.

The proposed amendment would also modify the TS for the applicable TS setpoints (i.e., the Average Power Range Monitors Simulated Thermal Power – Upscale trip setpoints) by adding requirements to assess channel performance during testing that verifies instrument channel setting values established by the plant-specific setpoint methodologies. This change is consistent with Option A of Technical Specifications Task Force Traveler, TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," (ML100060064) and Errata (ML101160026) made available in Reference 6.3. The proposed changes are also consistent with those made by similar Boiling Water Reactors (BWRs) in the implementation of Measurement Uncertainty Recapture (MUR) license amendments (References 6.4 through 6.6).

In addition, several TS references to certain percentages of RTP are being modified to reflect the increase in RTP.

2.0 Proposed Changes

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

Proposed changes to the Technical Requirements Manual (TRM) and TS Bases are also described below, with marked-up pages included in Enclosure 3. These changes are for information only, and do not require NRC approval.

2.1 Fermi 2 Operating License

Changes related to the value of RTP for Fermi 2, Facility Operating License Number NPF-43, Section 2.C.(1), "Maximum Power Level," are revised to increase the value of RTP from 3430 MWt to 3486 MWt.

2.3 Fermi 2 Technical Specifications Definitions

The definition of RTP in TS Section 1.1, "Definitions," is revised to increase the value of RTP from 3430 MWt to 3486 MWt.

2.4 Fermi 2 Technical Specifications 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"

Required Action E.1 in TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," associated with the applicable modes or other specified conditions for the Turbine Stop Valve – Closure and Turbine Control Valve – Fast Closure trip functions, is revised to change the value for RTP from 30% to 29.5%.

Surveillance Requirement (SR) 3.3.1.1.16 in TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," associated with the bypass of the Turbine Stop Valve – Closure and Turbine Control Valve – Fast Closure trip functions is revised to change the value for RTP from 30% to 29.5%.

Surveillance Requirement (SR) 3.3.1.1.20 in TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," associated with bypass of the Average Power Range Monitors Oscillation Power Range Monitor (OPRM) Upscale trip function is revised to change the value for RTP from 28% to 27.5%.

Technical Specifications Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b, Average Power Range Monitors Simulated Thermal Power – Upscale trip function Allowable Values (AVs) are revised as follows:

Allowable Value:
Current: ≤ 0.63 (W-ΔW) + 64.3% RTP and $\leq 115.5\%$ RTP
Proposed: ≤ 0.62 (W-ΔW) + 63.1% RTP and $\leq 115.5\%$ RTP

The following notes relating to instrument channel performance during testing in TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b, Average Power Range Monitors Simulated Thermal Power – Upscale are added for SR 3.3.1.1.18:

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

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

Technical Specifications Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Function 9, Turbine Stop Valve - Closure, and Function 10, Turbine Control Valve – Fast Closure, applicable modes or other specified conditions are revised to change the value for RTP from 30% to 29.5%.

2.5 Fermi 2 Technical Specifications 3.4.1, "Recirculation Loops Operating"

Technical Specifications Limiting Condition for Operation (LCO) 3.4.1 associated with single loop operation is revised to change the thermal power limits from ≤ 67.2% RTP to ≤ 66.1% RTP.

2.6 TRM Changes

Technical Requirements Manual Table TR3.3.1.1-1, "Reactor Protection System Instrumentation," Function 2.b.1, Average Power Range Monitors Simulated Thermal Power – Upscale Flow Biased trip setpoints are revised as follows:

Trip Setpoints:
Current: $\leq 0.63 (W-\Delta W) + 61.4\%$ RTP
Proposed: $\leq 0.62 (W-\Delta W) + 60.2\%$ RTP

In addition, the following note is added to this table for Function 2.b.1, describing the Fermi 2 instrument setpoint methodology used for this function:

(g) The method for determining the Nominal Trip Setpoints, as-found tolerances and as-left tolerances for this function are contained in Fermi 2 setpoint calculations. Setpoint calculations for this function are in accordance with the methods described in GEH Licensing Topical Reports NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996 and NEDE-33633P, "GEH Methodology for Implementing TSTF-493 Revision 4," February 2011.

Technical Requirements Manual Table TR3.3.2.1-1, "Control Rod Block Instrumentation," Function 3.a.1, Average Power Range Monitors Simulated Thermal Power – Upscale Flow Biased AVs are revised as follows:

Allowable Values:
Current: $\leq 0.63 (W-\Delta W) + 58.5\%$ RTP
Proposed: $\leq 0.62 (W-\Delta W) + 57.4\%$ RTP

Technical Requirements Manual Table TR3.3.2.1-2, "Control Rod Block Instrumentation," Function 3.a.1, Average Power Range Monitors Simulated Thermal Power – Upscale Flow Biased trip setpoints are revised as follows:

Trip Setpoints:
Current: $\leq$ 0.63 (W-ΔW) + 55.6% RTP
Proposed: $\leq$ 0.62 (W-ΔW) + 54.5% RTP

New TRM Section 3.3.7.3, "Feedwater Flow Instrumentation," is added to specify the proposed requirements for an inoperable LEFM system.

2.7 TS Bases Changes

The Bases for Section 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are changed to incorporate the revised values at which the Average Power Range Monitors OPRM Upscale, Turbine Stop Valve - Closure and Turbine Control Valve - Fast Closure trip functions are enabled.

The Bases for SR 3.3.1.1.16 in Section 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are changed to incorporate the revised value at which the Turbine Stop Valve - Closure and Turbine Control Valve - Fast Closure trip functions are enabled.

The Bases for SR 3.3.1.1.18 in Section 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are changed to incorporate a discussion of the footnotes regarding evaluation of instrument channel performance during testing.

The Bases for SR 3.3.1.1.20 in Section 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are changed to incorporate the revised value at which the Average Power Range Monitors OPRM Upscale trip function is enabled.

3.0 Technical Evaluation

3.1. Background and General Approach

10 CFR 50, Appendix K, paragraph I.A, "Sources of Heat During the LOCA," requires that emergency core cooling system (ECCS) evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change 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.

Utilization of the Cameron CheckPlus™ LEFM system at Fermi 2 has resulted in reduced uncertainty in feedwater flow measurement that reduces the total power level measurement uncertainty. As described in Section 3.2, "LEFM Flow Measurement and Core Thermal Power and Uncertainty," the core thermal power measurement uncertainty is a maximum of ±12.373 MWt (±0.355% of the MUR RTP or ±0.361% of the Current Licensed Thermal Power (CLTP)) with a fully functional LEFM system.

As summarized in Section 3.4.1, "Summary of Analyses," below and Enclosure 7, the ECCS evaluation models and other plant safety analyses currently assume an uncertainty of 2% of the CLTP (3430 MWt). Utilization of the LEFM system thus supports an increase in RTP to the requested 3486 MWt or approximately 1.64% of the CLTP. The sum of the requested RTP value (3486 MWt) and the maximum uncertainty value (12.373 MWt) is bounded by 102% of the CLTP value (3499 MWt) assumed in the plant safety analyses.

DTE has evaluated the effects of a 1.64% increase in RTP using an approach developed by General Electric-Hitachi (GEH) Nuclear Energy and approved by the NRC, which is documented in NEDC 32938P-A, "Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," (Reference 6.7). These evaluations are summarized in Section 3.4.1, "Summary of Analyses," and described in detail in Enclosure 7.

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

The proposed change would also modify the TS for the instrumentation with revised setpoints related to the power uprate. The proposed change formalizes new test requirements, thereby ensuring the instrument will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. This TS change is made through the addition of individual footnote requirements to the channel calibration surveillance for the Average Power Range Monitors Simulated Thermal Power – Upscale trip function.

3.2. LEFM Flow Measurement and Core Thermal Power Uncertainty

3.2.1 LEFM Flow Measurement

The LEFM system uses ultrasonic transit time principles to determine fluid velocity. This flow measurement method is described in topical reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System," Revision 0 (Reference 6.8) and ER-157 (P-A), "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or LEFM CheckPlus System," Revision 8 and Revision 8 Errata (Reference 6.9). These topical reports were approved by the NRC in documents titled, "Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report ER-80P, 'Improving Thermal Power Accuracy and Plant Safety

While Increasing Power Level Using the LEFM System,'" (Reference 6.10) and "Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, 'Caldon Ultrasonics Engineering Report ER-157P, Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System,' (TAC No. ME1321)," (Reference 6.11).

In References 6.10 and 6.11, the NRC established criteria for use of these topical reports in requests for license amendments. DTE's response to those criteria is provided in Section 3.2.4, "Disposition of NRC Criteria for Use of LEFM Topical Reports."

This instrumentation is not safety-related. However, it is subject to Fermi 2 augmented quality requirements (i.e., it is classified as QA1M). In addition, the LEFM system was designed and manufactured in accordance with Cameron's Quality Assurance Program. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM system are provided in Reference 6.8, Section 6.4 and Table 6.1.

3.2.2 Plant Implementation

The Fermi 2 LEFM was installed and commissioned in accordance with the appropriate Cameron installation and testing procedures. The LEFM spool pieces were installed in the feedwater piping of the two feedwater loops as shown in the installation drawings provided in Enclosure 15.

The installations in feedwater Loop A and Loop B are located in straight sections of 24 inch feedwater pipe upstream of the existing flow straighteners and feedwater flow venturis. In feedwater Loop A, the LEFM spool piece is installed 86 inches downstream of an elbow in the feedwater line (measured from the edge of the elbow to the centerline of the spool piece). In feedwater Loop B, the LEFM spool piece is installed 135 inches downstream of an elbow in the feedwater line (measured from the edge of the elbow to the centerline of the spool piece). Both spool pieces are located a distance of greater than three pipe diameters from a major hydraulic disturbance (e.g., piping elbow), as required by Cameron spool piece installation specifications.

It should be noted that in Loop B, the LEFM spool piece is installed 43 inches upstream of an elbow in the feedwater line (measured from the edge of the elbow to the centerline of the spool piece). This configuration does not have a significant effect on LEFM performance as discussed in the response to Criterion 7 in Section 3.2.4, "Disposition of NRC Criteria for Use of LEFM Topical Reports."

The transducers are located in the Turbine Building second floor mezzanine Main Steam Tunnel in a radiation field of approximately 750 mR/hr at full power, based on radiation survey data from the Fall of 2011. This would result in an exposure for the transducers of approximately 0.3 MRads over a 40 year plant life. The electronics cabinet is located on east side of the second floor of the Turbine Building in a radiation field of less than 1 mR/hr at full power. The material in the LEFM transducers has been exposed to gamma irradiation levels of 10 MRads with negligible degradation in transducer performance. Therefore, no radiation damage or degradation to the instruments due to the exposure levels seen in the plant is anticipated.

Following installation, testing included an inservice leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. Final commissioning testing is described in Appendix F of Reference 6.8.

3.2.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology

Enclosure 10 provides the results of testing and calibration of the LEFM system at Fermi 2. The Fermi 2 results indicate a feedwater mass flow rate uncertainty of ± 0.28% with a fully functional LEFM system. This uncertainty was calculated using the methodology described in Reference 6.9, which was approved by the NRC in Reference 6.11.

Based on a feedwater mass flow rate uncertainty of ± 0.28%, the core thermal power uncertainty calculation for Fermi 2 yields a total uncertainty of ± 12.373 MWt (±0.355% of MUR RTP or ±0.361% of CLTP) for the site-specific installation at a 95.5% confidence level. The calculation methodology is in accordance with the Fermi 2 setpoint calculation methodology. The calculation is provided in Enclosure 13 and provides further discussion of the uncertainty in the core thermal power, including the contributions of each parameter to the total core thermal power uncertainty.

3.2.4 Disposition of NRC Criteria for Use of LEFM Topical Reports

In References 6.10 and 6.11, the NRC established criteria to be addressed by licensees incorporating the LEFM methodology into the licensing basis. The criteria are listed below, along with a discussion of how each is or will be satisfied.

Criterion 1

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

Response to Criterion 1

Calibration and Maintenance

Implementation of the power uprate license amendment will include development of the necessary procedures and documents required for maintenance and calibration 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 CLTP of 3430 MWt. Initial preventive maintenance scope and frequency will be based on vendor recommendations

(See Enclosure 6, Item 3). The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

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

The LEFM system software and the plant process computer software configuration is maintained using existing Fermi 2 procedures, which include verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM system and the calorimetric process instrumentation is maintained is accordance with Fermi 2 configuration control procedures.

Fermi 2 programs and procedures addressing corrective actions, reporting deficiencies, and receiving and evaluating manufacturer's deficiency reports are discussed in Section 3.2.5, "Deficiencies and Corrective Actions."

LEFM Inoperability

The redundancy inherent in the two measurement planes of an LEFM system makes the system tolerant to component failures. Continuously operating online self-diagnostic testing is provided to verify that the digital circuits are operating correctly and within the design basis uncertainty limits, and LEFM system malfunctions result in Control Room alarms. The plant process computer also provides computer alarm messages in the Control Room if the status of the LEFM instrumentation changes. An out-of-specification condition will result in a self-diagnostic alarm condition, either for "Major Alert" status (i.e., increased flow measurement uncertainty), or "Fail" status. In these cases, the LEFM will be considered non-operational and the proposed Technical Requirements Manual (TRM) actions will be applied. Additionally, if the interface between the LEFM system and the plant process computer has failed, the LEFM will be considered non-operational and the proposed TRM actions will be applied.

As provided in Enclosure 3, the limitations discussed below regarding operation with an inoperable LEFM system will be included in the TRM, which will be revised prior to implementation (See Enclosure 6, Item 1).

The proposed TRM specification requires a LEFM channel check every 12 hours. In addition to this confirmation of status, the Control Room alarms and plant process computer alarm messages described above alert the operators if the status of the LEFM instrumentation changes.

A process will be implemented to use the LEFM system feedwater flow to adjust or correct the existing feedwater flow venturi-based signals (See Enclosure 6, Item 2). If the LEFM system or a portion of the system becomes non-operational as discussed above, Control Room operators are promptly alerted by a Control Room alarm. Feedwater flow input to the core thermal power calculation would then be provided by the existing feedwater flow venturis.

Since the feedwater flow venturis will be corrected to the last validated data from the LEFM system, it is acceptable to remain at the uprated RTP of 3486 MWt for up to 72 hours to enact LEFM system repairs. As noted in the TRM changes provided, if the LEFM system is not repaired within 72 hours, power will be reduced and administratively controlled to remain less than or equal to the CLTP of 3430 MWt.

The 72 hour Completion Time for the LEFM system prior to reducing to the CLTP is acceptable. As discussed above, during the 72 hour Completion Time, the existing feedwater flow venturi-based signals will be corrected to the last validated data from the LEFM system. Although the feedwater flow venturi measurements may drift slightly during this period due to fouling, fouling of the feedwater flow venturis results in a higher than actual indication of feedwater flow. This condition results in an overestimation of the calculated calorimetric power level, which is conservative, as the reactor will actually be operating below the calculated power level. Note that the NRC has previously approved power uprate applications with Completion Times of up to 72 hours for similar BWRs (References 6.4 through 6.6).

A sudden de-fouling event during the 72 hour inoperability period is unlikely. Significant sudden de-fouling would be detected by a change in the balance of plant parameters. A review of recent plant operating experience has not identified any instances of sudden de-fouling events at Fermi 2.

Regarding potential drift in the measurement of feedwater differential pressure across the feedwater flow venturis, industry experience for similar BWRs shows that the instrument drift associated with feedwater flow measurements are insignificant over a 72 hour time period. Differences in the feedwater loop flow rates measured by the feedwater flow venturis and the LEFM were compared for the time period since LEFM commissioning. Evaluation of the data indicates a maximum change in the difference between the feedwater flow venturi and the LEFM measured flow rates of approximately 0.3% over an 18 month operating cycle (or less than 0.002% over a 72 hour period). Thus, the effects of instrument drift and/or fouling of the feedwater flow venturis would be insignificant during the time period the LEFM is allowed to be out of service.

If the core power level is below the CLTP at the time the LEFM is declared non-operational or if the power level drops below the current CLTP during the Completion Time, power may not be raised above 3430 MWt prior to the LEFM returning to operational status. In Section 3.0 of the Fermi 2 TRM, Technical Requirements Limiting Condition for Operation (TRLCO) 3.0.4 prohibits entering a condition specified in the applicability when a TRLCO is not met, except when either (a) the associated actions permit operation in that condition for an unlimited period of time, or (b) upon acceptable performance of a risk assessment and establishment of appropriate risk management actions. Exception (a) cannot be used for the LEFM, since the applicability for proposed LEFM TRLCO applies to core thermal power levels greater than 3430

MWt and the TRM actions only permit operation above 3430 MWt for 72 hours with a non-operational LEFM. Regarding exception (b), the proposed Fermi 2 TRM section includes a note stating that TRLCO 3.0.4.b does not apply to the LEFM. Thus, the application of TRLCO 3.0.4 to the proposed TRM section for the LEFM would prohibit raising power above 3430 MWt without the LEFM being operational.

In the event that the plant process computer is inoperable, the LEFM would be considered non-operational and the proposed TRM actions will be applied, as described above. A procedure currently exists for reactor engineering personnel to manually calculate core thermal power. In addition, operators routinely monitor other indications of core thermal power, including Average Power Range Monitors (APRMs), steam flow, feedwater flow, turbine first stage pressure, and main generator output.

Criterion 2

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

Response to Criterion 2

The LEFM system was installed at Fermi 2 during the Fall 2010 refueling outage and was commissioned in February 2011. The LEFM system is being used to supply the feedwater flow input to the plant process computer core thermal power calculation. Since the commissioning of the LEFM, the following maintenance issues have occurred:

- In September 2011, the LEFM inputs to the plant computer system failed. Feedwater flow inputs to the core thermal power calculation were transferred to the feedwater flow venturis. The LEFM CPU was rebooted. Post reboot data reports were analyzed by Cameron and they determined that the LEFM was functioning as designed and within the commissioning uncertainty bounds after the reboot.

- In November 2011, during a plant down power, a “FW LEFM Meter B Status - Major Alert” alarm was received and the alarm cleared at approximately 23% of CLTP during the subsequent power increase. The cause of the alarm was determined to be a failed RTD in feedwater Meter B Plane 3. This failed component does not adversely effect the operation of the LEFM and the RTD is currently scheduled to be replaced during the next refueling outage.

- Also in November 2011, a small leak was discovered in the tubing for the feedwater Loop A pressure instrument. Feedwater flow inputs to the core thermal power calculation were transferred to the feedwater flow venturis, the pressure instrument was isolated, and the tubing leak was repaired.

- In February 2012, a number of LEFM Meter B Plane 3 computer alarms were received. These alarms did not adversely effect the operation of the LEFM. Troubleshooting was performed by Cameron and an adjustment was made to the "K_{max}" factor for Meter B Plane 3. Post maintenance data reports were analyzed by Cameron and they determined that the LEFM was functioning as designed following the maintenance.

As was mentioned above, final commissioning of the LEFM system at Fermi 2 was completed on February 24, 2011. The commissioning process verified bounding calibration test data, as described in Appendix F of Reference 6.8. This step provided final confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Enclosure 10.

Criterion 3

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

Response to Criterion 3

The LEFM system uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) PTC 19.1-1985 and the Instrumentation, Systems, and Automation Society (ISA) RP67.04.02-2000 methodologies, as described in Enclosure 10. This LEFM system uncertainty calculation methodology is based on a square-root-sum-of-squares (SRSS) calculation, as described in Reference 6.9.

The Fermi 2 core thermal power uncertainty calculation for the LEFM feedwater flow instrumentation (Enclosure 13) was done in accordance with the Fermi 2 instrument setpoint methodology. The core thermal power uncertainty calculation for the existing feedwater flow instrumentation was also done in accordance with the Fermi 2 instrument setpoint methodology, and thus used consistent methodology.

Criterion 4

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

Response to Criterion 4

Criterion 4 does not apply to Fermi 2. The calibration factors for the Fermi 2 spool pieces were established by tests of these spools at Alden Research Laboratory. These tests were performed on a full-scale model of the Fermi 2 hydraulic geometry. A discussion of the impact of the plant-specific installation factors on the feedwater flow measurement uncertainty is provided in Cameron Report ER-781, Revision 2 (Enclosure 10) and Cameron Report ER-818, Revision 0 (Enclosure 11). The test configurations modeled the portion of piping upstream of the LEFM spool pieces. The tested configuration of the LEFM spool pieces can be compared to the plant installation drawings by comparing the drawings in ER-818, Figures 1 and 2, to the installation drawings in Enclosure 15. There is no significant difference between the Fermi 2 feedwater piping configuration and the model used at Alden Research Laboratory.

As was discussed above, the commissioning process for the Fermi 2 LEFM was completed by Cameron on February 24, 2011.

Criterion 5

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

Response to Criterion 5

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

Criterion 6

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

Response to Criterion 6

Fermi 2 will not consider the CheckPlus system with a single failure as a separate category. In these cases, the CheckPlus LEFM system will be considered as inoperable and the actions identified in the response to Criterion 1 above will be implemented.

Criterion 7

An applicant with a comparable geometry can reference the Section 3.2.1 finding (of Reference 6.11) to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.

Response to Criterion 7

The configuration of the LEFM spool pieces is described in Section 3.2.2, "Plant Implementation." The spool piece configuration for feedwater Loop B is comparable to that described in Section 3.2.1 of Reference 6.11. Based on conducting tests at Alden Laboratories described in Enclosure 11 and the NRC findings in Section 3.2.1 of Reference 6.11, the downstream geometry does not have a significant influence on the Fermi 2 LEFM calibration.

As is discussed in the response to Criterion 6 above, Fermi 2 will not consider the CheckPlus system with a single failure as a separate category and the CheckPlus LEFM system will be considered as inoperable.

Criterion 8

An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 (of Reference 6.11) should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 (of Reference 6.11). Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.

Response to Criterion 8

Fermi 2 does not have flow straighteners upstream of the LEFM spool piece installations. Thus, this criterion is not applicable to Fermi 2.

Criterion 9

An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 (of Reference 6.11).

Response to Criterion 9

Fermi 2 conservatively assumes no moisture content in the Core Thermal Power Uncertainty Calculation (Enclosure 13). This approach is consistent with that described in Section 3.2.3 of Reference 6.11. Thus, this criterion is not applicable to Fermi 2.

3.2.5 Deficiencies and Corrective Actions

Cameron has procedures to notify users of important LEFM deficiencies. Fermi 2 also has processes for addressing manufacturer's deficiency reports. Such deficiencies are documented in the Fermi 2 corrective action program.

Problems with plant instrumentation identified by Fermi 2 personnel are also documented in the Fermi 2 corrective action program and necessary corrective actions are identified and implemented. Deficiencies associated with the vendor's processes or equipment are reported to the vendor to support corrective action.

3.2.6 Reactor Power Monitoring

Fermi 2 procedure MOP03, "Policies and Practices," provides guidance to ensure that reactor power remains within the requirements of the operating license. Procedure Section 3.10 provides guidance for monitoring and controlling reactor power that is consistent with the guidance proposed by the Nuclear Energy Institute and endorsed by the NRC in Reference 6.12.

3.3. Evaluation of Changes to Operating License and Technical Specifications

The proposed changes to the TS described in Section 2.0, "Proposed Changes," are evaluated below. The numbering of these changes corresponds to the numbering in Section 2.0.

Sections 2.1 and 2.2, Change in RTP

The proposed increase of approximately 1.64% in RTP in the Fermi 2 Operating License and TS Definitions is acceptable based on the decreased uncertainty in the core thermal power calculation due to the use of the LEFM system and on the evaluations provided in this License Amendment Request.

Section 2.3, Revised Bypass Values for Turbine Stop Valve - Closure and Turbine Control Valve - Fast Closure Trip Functions

The proposed change for the power level at which the Turbine Stop Valve - Closure and Turbine Control Valve - Fast Closure trip functions are bypassed are contained in TS 3.3.1.1 Required Action E.1, SR 3.3.1.1.16, and Table 3.3.1.1-1, Functions 9 and 10. The bypass of these trip functions is accomplished by sensing turbine first-stage pressure. Based on the guidelines in Reference 6.7, Section F.4.2.3, "Turbine First-Stage Pressure Signal Setpoint," the value at which the Turbine Stop Valve - Closure trip and Turbine Control Valve - Fast Closure trip functions are bypassed, in percent of RTP, is reduced by the ratio of the power increase. The value does not change with respect to absolute thermal power.

Section 2.3, Revised Bypass Value for Average Power Range Monitors OPRM Upscale Trip Function

The proposed change for the power level at which the Average Power Range Monitors OPRM Upscale trip function is bypassed is contained in TS 3.3.1.1, SR 3.3.1.1.20. Fermi 2 is operating under the requirements of reactor stability Long-Term Solution Option III. The Option III solution monitors OPRM signals to determine when a reactor scram is required. The OPRM system will only cause a scram when plant operation is in the Option III armed region. For MUR operation, the armed region is rescaled to maintain the pre-MUR absolute thermal power, and thus the power level expressed in percent RTP decreases in proportion to the power uprate.

Section 2.3, Revised Allowable Values for the Average Power Range Monitors Simulated Thermal Power – Upscale Trip Function

The proposed changes in the two-loop and single loop Average Power Range Monitors Simulated Thermal Power - Upscale trip functions are contained in TS 3.3.1.1 Table 3.3.1.1-1. The proposed change to the Allowable Values (AVs) for the Average Power Range Monitors Simulated Thermal Power - Upscale trip functions are based on the approach described in Reference 6.7, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." The Average Power Range Monitors Simulated Thermal Power – Upscale trip functions Analytical Limits (ALs) and AVs, for both two-loop operation and single loop operation, are unchanged in units of absolute core thermal power versus recirculation drive flow. Because these values are expressed in percent of RTP, they decrease in proportion to the power uprate. The specific values for the ALs are provided in Enclosure 7, Section 5.3, "Technical Specification Instrument Setpoints." The AVs were generated using standard GEH setpoint methodologies. The Fermi 2 instrument setpoint calculations are provided in Enclosure 14. Further discussion of the setpoint methodology is found in this document in Section 3.4.4, "Instrument Setpoint Methodology."

Section 2.3, Changes Related to Instrument Channel Performance during Testing

A discussion of these changes is provided in Section 3.4.4, "Instrument Setpoint Methodology."

Section 2.4, Revised Power Limit for Single Loop Operation

The proposed changes to the thermal power limit for single loop operation is contained in TS 3.4.1. The proposed change to the power limitation for single loop operation is based on the approach described in Reference 6.7, Section 5.2, "Power/Flow Map." The limiting value is unchanged in units of absolute core thermal power. Because this value is expressed in percent of RTP, it decreases in proportion to the power uprate.

3.4. Additional Considerations

3.4.1 Summary of Analyses

The following is a summary of the analyses performed in support of these proposed changes, along with the results and a reference to the sections of Enclosure 7 providing further detail.

Topic	Conclusion	Enclosure 7 Section
Normal plant operating conditions	Uprate accommodated within previously licensed power-flow map.	Section 1
Reactor core and fuel performance	All fuel and core design limits met.	Section 2
Reactor coolant and connected systems	Overpressure protection, fracture toughness, structural, and piping evaluations acceptable.	Section 3
Engineered safety features	Acceptable based on previous analyses at 102% of current licensed power.	Section 4
Instrumentation and control	Current instrumentation acceptable; changes to some TS and TRM values.	Section 5
Electrical power and auxiliary systems	Minor increases in normal power system loads; emergency power systems unaffected; auxiliary systems acceptable.	Section 6
Power conversion systems	Power conversion systems adequate without modification.	Section 7
Radwaste and radiation sources	Small increases in normal operation radiation levels and effluents; accident consequences bounded by previous evaluations.	Section 8
Reactor safety performance evaluations	Design basis events bounded by previous evaluations, special events meet acceptance criteria.	Section 9
Other evaluations	All evaluation results acceptable.	Section 10

3.4.2 Adverse Flow Effects

Industry experience has revealed that power uprate conditions can cause vibrations associated with acoustic resonance that can lead to steam dryer and main steam line (MSL) valve degradation. This experience has been associated with extended power uprates (EPUs), and not with smaller uprates, such as stretch or MUR uprates.

Fermi 2 has performed steam dryer baseline examinations in accordance with Boiling Water Reactor Vessel Internals Project (BWRVIP)-139, “BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines,” April 2005. Re-examinations of the steam dryer welds and locations are being performed in accordance with BWRVIP-139-A, “BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines,” July 2009. No changes to the steam dryer examination program are necessary for implementation of the MUR uprate. Regarding steam dryer flow induced vibration at MUR uprate conditions, an independent analysis was performed using vibration data from Fermi 2 and a similar plant that has implemented an EPU. The results of the analysis indicate that although steam dryer loads and stresses increase slightly due to the MUR uprate conditions, they remain within allowable limits. Thus, implementation of an MUR uprate at Fermi 2 poses a small risk to the structural integrity of the steam dryer.

Independent 1/8th scale model testing was performed on a model of the Fermi 2 main steam lines from the reactor vessel to the main turbine to determine the effect the MUR uprate might have on flow induced vibration of the piping near the turbine control valves. The piping pressure fluctuations and resulting vibration were previously determined to be a function of the valve lift settings of the turbine control valves. The scale model testing demonstrated that there is a peak pressure fluctuation that occurs when the valve lift is approximately 85%, which is above the current plant limit and above the expected valve position at the MUR uprated conditions.

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

3.4.3 Plant Modifications

The evaluations performed to support the power uprate identified that modifications are required to certain systems, such as minor equipment changes or replacements (e.g., feedwater heater relief valves), and setpoint or alarm point changes (e.g., TS instrument setpoint changes). These modifications will be made in accordance with the requirements of 10 CFR 50.59, “Changes, Tests, and Experiments,” and will be implemented prior to, or concurrently with, implementation of the proposed power uprate (See Enclosure 6, Item 4).

3.4.4 Instrument Setpoint Methodology

As described in Section 2.0, “Proposed Changes,” the only proposed change to TS Limiting Safety System Setpoints (LSSSs) is for the Average Power Range Monitors Simulated Thermal Power - Upscale trip function. The Nominal Trip Setpoints and AVs for this function were generated using the simplified GEH setpoint methodology described in Section 5.3.3 of NEDC-33004P-A, “Constant Pressure Power Uprate,” July 2003. As required by Reference 6.3, the Fermi 2 setpoint calculation for the Average Power Range Monitors Simulated Thermal Power - Upscale trip function is included in Enclosure 14.

Consistent with Option A of Technical Specifications Task Force Traveler, TSTF-493, Revision 4, "Clarify Application of Setpoint Methodology for LSSS Functions," and Errata, made available in Reference 6.3, the Average Power Range Monitors Simulated Thermal Power - Upscale trip function is to be included in functions requiring TS SR controls to provide adequate assurance that instruments will actuate safety functions at the point assumed in the applicable safety analysis. Thus, the TSTF-493, Revision 4, Option A footnotes described in Section 2.3 are applied to the SR for channel calibration for this function (SR 3.3.1.1.18). Discussion of the notes and the methodology for determining the as-found and as-left tolerances is added to the TRM and TS Bases associated with this function. The associated TRM and TS Bases changes are included in Enclosure 3. Plant procedures will ensure that the requirements of the TSTF-493, Revision 4, Option A footnotes are implemented (See Enclosure 6, Item 5).

As discussed in TSTF-493, Revision 4, the as-found and as-left tolerances described in the footnotes for SR 3.3.1.1.18 are not applied to digital components. Fermi 2 uses a digital Nuclear Measurement Analysis and Control (NUMAC) based Power Range Neutron Monitoring (PRNM) system to perform the Average Power Range Monitors Simulated Thermal Power – Upscale trip function. Analog recirculation loop drive flow inputs are used to digitally determine the setpoints for this function. As such, the as-found and as-left tolerances will only be applied to the analog recirculation loop drive flow inputs in the channel calibration procedure for this function.

3.4.5 Grid Studies

A grid adequacy study was performed to support Fermi 2 MUR and used the methodology for periodic (annual) grid studies specified in the Fermi 2 Nuclear Plant Operating Agreement (NPOA). This study used a current forecasted system configuration, and included the current forecasted loads and generation dispatch. The Fermi 2 plant was dispatched at 1215 MWe output, the nameplate electrical rating of the Fermi 2 main generator, which bounds the projected generator output after MUR.

The Fermi 2 off site power system consists of two off site sources. System Service Transformer 65 (SS#65), fed from 345 kV Bus 301, supplies the plant Division 2 loads at 4.16 kV and System Service Transformer 64 (SS#64), fed from 120 kV Bus 101, supplies the plant Division 1 loads at 4.16 kV.

Operation of the grid and Fermi 2 off site power system is governed by a NPOA between DTE, the International Transmission Company (ITC) and the Midwest Independent Transmission System Operator (MISO). The NPOA is required by NERC Standard NUC-001-2 and contains the Nuclear Plant Interface Requirements (NPIRs) for Fermi 2. The current NPOA (Revision 6) contains the following voltage and voltage drop limits for Fermi 2:

345 kV Bus Voltage Limits
  Low Limit – 98.4% of nominal 345 kV (339.5 kV)
  High Limit – 105% of nominal 345 kV (362 kV)

120 kV Bus Voltage Limits
Low Limit – 93.3% of nominal 120 kV (112 kV)
High Limit – 105% of nominal 120 kV (126 kV)
345 kV Bus Voltage Drop Limit – 2.7%
120 kV Bus Voltage Drop Limit – 1.6%

The program used to perform this study was the Siemens PTI Power System Simulator for Engineering (PSS/E). This program is used by many of the transmission owners on the Eastern Interconnection of the United States. The PSS/E program includes modules that perform both steady state and transient analysis of a transmission system.

The cases used for this study were modeled to simulate load levels and generation dispatch expected during 2012. Six cases were modeled for this study, and are described below. Summer load cases were used, since the area transmission system is a summer peaking system (i.e., the summer peak and off peak analyses performed bounds the winter and shoulder load periods).

1. A 2012 summer peak load case was modeled with the expected full generation dispatch (Case 100N).
2. A 2012 summer peak load case with several nearby generators modeled out of service to stress the voltage conditions in the area of the Fermi site (Case 100S)
3. A 2012 summer peak load case with Fermi 2 out of service and the other nearby generators fully dispatched (Case 100EF).
4. A 2012 summer peak load case with a design minimum degraded grid voltage, with targeted voltages of 95% at 345 kV and 120 kV busses (Case 100LV).
5. An 80% 2012 summer peak load (conforming load scaled down only) ITC Transmission (ITCT)/Michigan Electric Transmission Company (METC) load case, with an economic order generation reduction (Case 80N).
6. An 80% 2012 summer peak load case with a design maximum grid high voltage, with targeted voltages of 105% at 345 kV and 120 kV busses (Case 80HV).

For this study, the planning criteria faults were modeled at both the Fermi 2 345 kV and 120 kV switchyards, as required by the NPOA. In addition, faults at the Brownstown 345 kV substation were simulated. The Brownstown substation was included as it is the only substation to which the Fermi 345 kV switchyard connects, and thus greatly influences the transient stability of Fermi 2. A total of 66 fault scenarios were modeled which included:

1. Three phase to ground faults with normal clearing.
2. Simultaneous one phase to ground faults on adjacent circuits on same tower with normal clearing.
3. Two phase to ground faults with delayed clearing (stuck circuit breakers or primary relay protection out of service).
4. With one circuit initially out of service, three phase faults with normal clearing on another circuit.
5. One phase to ground faults on circuit breakers with normal clearing.

6. Additional simulations were conducted to analyze the trip of the Fermi 2 generator without fault.

Each of the 66 fault scenarios were run for the six cases described above (for a total of 396 fault simulations). For each fault simulation, mechanical power, real and imaginary electrical power, machine terminal voltage and machine speed deviation responses were plotted for the Fermi 2 generator and eight additional generators in the area which were expected to have the greatest interaction with Fermi 2. Each fault simulation was carried out for 10 seconds, tripping all lines, buses, transformers, loads and generators at the appropriate time. If it was not clear that a well damped, stable response had been obtained after 10 seconds, the simulation of that particular fault was carried out further until a stable response was obtained. Each fault simulation demonstrated a stable, well damped response. None of the fault simulations led to transient instability at Fermi 2 or any of the other area generators monitored.

A voltage analysis was performed for the six cases described above. Voltage was monitored on the Fermi 2 system service buses SS#64 and SS#65 (4.16kV buses) and the grid transmission buses (345 kV and 120 kV) during each of the fault simulations. The PSS/E program was executed to record the needed voltages once every 1.25 cycles for the entire 10 second simulation. The voltage and time for under voltage relay reset was determined and included in this study, a voltage profile over time with application of the under voltage relay logic was used to determine simulated relay flags, resets and tripping due to transient and/or steady state voltage conditions. The pre-fault voltage, the minimum voltage seen during the fault itself, and the steady state (at 10 seconds) voltage were noted for each fault simulation. A percentage voltage drop for both the 120 kV and 345 kV buses was calculated based on the initial and steady state simulation voltage.

For the simulations where a trip of the Fermi 2 generator without fault occurs (with the exceptions of Case 100EF where Fermi 2 is off line and Case 100LV where initial grid voltage is below that required by the NPIRs), minimum voltages for the SS#64 and SS#65 4.16 kV busses were above the minimum voltages specified in the NPIRs and voltage drops for these busses were also within the NPIR limits.

For the simulations where the fault causes the loss of either the SS#64 or SS#65 transformer (with the exception of Case 100LV where initial grid voltage is below that required by the NPIRs), the minimum voltage and voltage drop for the remaining 4.16 kV bus were within the NPIR limits.

The results show that the Fermi 2 generator and the rest of the transmission system will remain stable for all conditions studied. Fermi 2 and all generators in the study area show well-damped stable responses to all faults simulated. The voltage at the Fermi 2 120 kV and 345 kV buses, as well as at the Fermi 2 critical system service 4.16kV buses, are sufficient to prevent the initiation of a trip of the degraded grid relays following a trip of the Fermi 2 generator. The grid is capable of supplying the required off site power if the Fermi 2 unit trips off line, which will allow safe shutdown during a Design Basis Accident (DBA) condition. Therefore, in accordance with 10

CFR 50, Appendix A, General Design Criteria 17, the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the Fermi 2 generator or the loss of power from the transmission network is minimized.

3.4.6 Operator Training, Human Factors, and Procedures

Operator response to transients, accidents, and special events is unaffected by the proposed changes. Necessary operating procedure revisions (including Emergency Operating Procedures and Abnormal Operating Procedures) will be completed prior to implementation of the proposed changes (See Enclosure 6, Item 5). The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes (See Enclosure 6, Item 6). Operator training will be completed prior to implementation of the proposed changes (See Enclosure 6, Item 7)).

3.4.7 Testing

Plant testing for the proposed changes will be completed as described in Enclosure 7, Section 10.4, “Testing,” (See Enclosure 6, Item 8).

4.0 Regulatory Evaluation

4.1. Applicable Regulatory Requirements/Criteria

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

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

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

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

4.2. Precedent

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

Facility	Amendment No(s).	Approval Date	Accession No.
Cooper Nuclear Station	231	June 30, 2008	ML081540280
Davis-Besse Nuclear Power Station	278	June 30, 2008	ML081410652
Calvert Cliffs, Units 1 and 2	291/267	July 22, 2009	ML091820366
North Anna, Units 1 and 2	257/238	October 22, 2009	ML092250616
LaSalle, Units 1 and 2	198/185	September 16, 2010	ML101830361
Limerick, Units 1 and 2	201/163	April 8, 2011	ML110691095

4.3. No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit" and 10 CFR 50, Appendix K, "ECCS Evaluation Models," the DTE Electric Company (DTE) requests an amendment to Facility Operating License No. NPF-43 for Fermi 2. Specifically, the proposed changes revise the Operating License and Technical Specifications (TS) to implement an increase of approximately 1.64% in RTP from 3430 megawatts thermal (MWt) to 3486 MWt. These changes are based on increased feedwater flow measurement accuracy, which was achieved by utilizing Cameron International (formerly Caldon) CheckPlus™ Leading Edge Flow Meter (LEFM) ultrasonic flow measurement instrumentation.

The proposed changes also revise the TS by adding test requirements to the TS instrument function affected by the power uprate to ensure that the instrument function will actuate as required to initiate protective systems at the point assumed in the applicable safety analysis.

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

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or
(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
(3) Involve a significant reduction in a margin of safety.

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

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

   Response: No

   The reviews and evaluations performed to support the proposed uprated power conditions included all components and systems that would be affected by the proposed changes. All accident mitigation systems will function as designed, and all performance requirements for these systems have been evaluated and were found acceptable. Thus, the proposed changes do not create any new accident initiators or increase the probability of an accident previously evaluated.

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

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

   A majority of the current safety analyses remain applicable, since they were performed at power levels that bound operation at a core power of 3486 MWt. Other analyses previously performed at the current licensed thermal power level have either been evaluated or re-performed for the increased power level. The results demonstrate that acceptance criteria of the applicable analyses continue to be met at the uprated conditions. As such, all applicable accident analyses continue to comply with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition.

   The proposed changes add test requirements to the revised TS instrument function related to variables that have a significant safety function to ensure that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. Surveillance tests are not an initiator to any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased. The systems and components required by the TSs for which surveillance test requirements are added are still required to be operable, meet the acceptance criteria for the surveillance requirements, and be capable of performing any mitigation function assumed in the accident analysis.

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

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

   Response: No

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

   The proposed changes to surveillance test requirements for the revised TS instrument function involve a physical alteration of the plant, i.e., a change in an instrument setpoint, but do not involve installation of a new or different type of equipment. The proposed changes do not alter assumptions made in the safety analysis but ensures that the instruments perform as assumed in the accident analysis. The proposed changes are consistent with the safety analysis assumptions.

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

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

   Response: No

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

   The proposed changes add test requirements to the revised TS instrument function that (1) will assure that TS instrumentation Allowable Values will be limiting settings for assessing instrument channel operability, and (2) will be conservatively determined so that evaluation of instrument performance history and the As Left Tolerance requirements of the calibration procedures will not have an adverse effect on equipment operability. The testing methods and acceptance criteria for systems, structures, and components, specified in applicable codes and standards (or alternatives approved for use by the Nuclear Regulatory Commission) will continue to be met as described in the plant licensing basis including the updated Final Safety Analysis Report. There is no impact to safety analysis acceptance criteria as described in the plant licensing basis because no change is made to the accident analysis assumptions.

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

4.4. Conclusions

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

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

5.0 Environmental Consideration

10 CFR 51.22, “Criterion for Categorical Exclusion; Identification of Licensing and Regulatory Actions Eligible for Categorical Exclusions or Otherwise Not Requiring Environmental Review,” addresses requirements for submitting environmental assessments as part of licensing actions. 10 CFR 51.22, paragraph (c)(9) states that a categorical exclusion applies for Part 50 license amendments that meet the following criteria:

i. No significant hazards consideration (as defined in 10 CFR 50.92(c));
ii. No significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and
iii. No significant increase in individual or cumulative occupational radiation exposure.

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

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

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

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

6.0 References

6.1 Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

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

6.3 Federal Register, Volume 75, No. 90, Page 26294 (75 FR 26294), “Notice of Availability of the Models for Plant-Specific Adoption of Technical Specifications Task Force Traveler TSTF-493, Revision 4, ‘Clarify Application of Setpoint Methodology for LSSS Functions,’” dated May 11, 2010.

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

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

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

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

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

6.9 Caldon Topical Report ER-157 (P-A), "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," Revision 8 and Revision 8 Errata, dated May 2008.

6.10 Letter from John N. Hannon (USNRC) to C. Lance Terry (TU Electric), "Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System,' (TACS Nos. MA2298 and MA2299)," dated March 8, 1999 (9903190053).

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

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

Enclosure 2
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Markup of Existing Operating License and Technical Specifications

8 Pages

(4) DECo, 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 such 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) DECo, 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) DECo, 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 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> 3486

DECo is authorized to operate the facility at reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Antitrust Conditions</u>

DECo shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between Detroit Edison Company and

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3430~~ 3486 MWt.

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

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

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

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

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. ----------NOTE--------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, and 2.f. --------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. <u>OR</u> B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~30~~% RTP. 29.5	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.12	-----NOTE----- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.14	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure Functions are not bypassed when THERMAL POWER is ≥ ~~30~~% RTP. 29.5	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.17	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	------------------NOTE------------------ Neutron detectors are excluded. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.19	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.20	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ ~~28~~% and recirculation drive flow is < 60% of rated recirculation drive flow. 27.5	24 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux – Upscale (Setdown)	2	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. Simulated Thermal Power – Upscale	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18 (d)(e)	62 63.1 ≤ 0.63 (W - ΔW) + ~~64.3~~% RTP and ≤ 115.5% RTP(b)

(continued)

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

(b) ΔW = 8% when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." Otherwise ΔW = 0%.

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

(d) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The NTSP and the methodologies used to determine the as-found and as-left tolerances are specified in the Technical Requirements Manual.

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Scram Discharge Volume Water Level - High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft. 0 inches
	5(a)	2	I	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft. 0 inches
b. Float Switch	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft. 0 inches
	5(a)	2	I	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft. 0 inches
9. Turbine Stop Valve - Closure	29.5 ≥ 30% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 7% closed
10. Turbine Control Valve Fast Closure	≥ 30% RTP 29.5	2	E	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
11. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.13 SR 3.3.1.1.15	NA
12. Manual Scram	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

Two recirculation loops with matched recirculation loop jet pump flows shall be in operation;

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;
3. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power-Upscale) Allowable Value of Table 3.3.1.1-1 is reset for single loop operation, when in MODE 1; and
4. THERMAL POWER is ≤ ~~67.2%~~ 66.1 RTP.

------------------------------NOTE------------------------------
Application of the required limitations for single loop operation may be delayed for up to 4 hours after transition from two recirculation loop operations to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

Enclosure 3
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Markup of Existing Technical Requirements Manual and Technical Specifications Bases

Technical Requirements Manual: 7 Pages
Technical Specifications Bases: 7 Pages

Technical Requirements Manual Markup

TR 3.3 INSTRUMENTATION

TR 3.3.1.1 Reactor Protection System (RPS) Instrumentation

The RPS instrumentation trip setpoints and response times are listed in Table TR3.3.1.1-1.

TABLE TR3.3.1.1-1 (Page 1 of 2)
Reactor Protection System Instrumentation

FUNCTION	TRIP SETPOINT	RESPONSE TIME (seconds)
1. Intermediate Range Monitors		
a. Neutron Flux - High	≤ 120/125 divisions of full scale	NA
b. Inop	NA	NA
2. Average Power Range Monitors[a]		
a. Neutron Flux-Upscale (Setdown)	≤ 15% RTP	NA
b. Simulated Thermal Power - Upscale	0.62 60.2%	NA
1. Flow Biased (g)	≤ 0.63 (W-ΔW)[b] + 61.4%,	
2. High Flow Clamped	with a maximum of ≤ 113.5% of RTP	
c. Neutron Flux - Upscale	≤ 118% RTP	NA
d. Inop	NA	NA
e. 2-out-of-4 Voters	NA	≤ 0.05[a]
f. OPRM-Upscale		NA
1. Confirmation Count and	14	
2. Amplitude	1.11	
3. Growth	1.3	
4. Amplitude	1.3	

(continued)

(a) Neutron detectors, APRM channel, and 2-out-of-4 Trip Voter digital electronics are exempt from response time testing. Response time shall be measured from activation of the 2-out-of-4 Trip Voter output relay.

(b) ΔW = 0% for two loop operation. ΔW = 8% for single loop operation.

TABLE TR3.3.1.1-1 (Page 2 of 2)
Reactor Protection System Instrumentation

	FUNCTION	TRIP SETPOINT	RESPONSE TIME (seconds)
3.	Reactor Vessel Steam Dome Pressure - High	≤ 1093 psig	≤ 0.55[c]
4.	Reactor Vessel Water Level - Low, Level 3	≥ 173.4 inches[d]	≤ 1.05[c]
5.	Main Steam Isolation Valve - Closure	≤ 8% closed	≤ 0.06
6.	Main Steam Line Radiation - High	≤ 3.0 x full power background[f]	NA
7.	Drywell Pressure - High	≤ 1.68 psig	NA
8.	Scram Discharge Volume Water Level - High		
	a. Level Transmitter	≤ 592 ft. 6 inches	NA
	b. Float Switch	≤ 594 ft. 8 inches	NA
9.	Turbine Stop Valve-Closure	≤ 5% closed	≤ 0.06
10.	Turbine Control Valve Fast Closure	Initiation of fast closure	≤ 0.08[e]

(c) The sensor and relays/logic response time need not be measured and may be assumed to be the design response time. Prior to return to service of a new transmitter/relay or following refurbishment of a transmitter (e.g., sensor cell or variable damper components/relay), a response time test will be performed to determine an initial sensor/relay specific response time value.

(d) As referenced to instrument zero Top of Active Fuel (TAF).

(e) Measured from de-energization of K37 relay, which inputs the turbine control valve closure signal, to the RPS.

(f) A new "full power background" level is established for hydrogen water chemistry based on 100% power operation with the established hydrogen injection rate. Actual background radiation levels may be less depending on actual power level or hydrogen injection rate.

Setpoint adjustment is not necessary for variations in power or hydrogen injection rate including interruptions in hydrogen flow.

(g) The method for determining the Nominal Trip Setpoints, as-found tolerances and as-left tolerances for this function are contained in Fermi 2 setpoint calculations. Setpoint calculations for this function are in accordance with the methods described in GEH Licensing Topical Reports NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996 and NEDE-33633P, "GEH Methodology for Implementing TSTF-493 Revision 4," February 2011.

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TABLE TR3.3.2.1-1 (Page 2 of 3)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Intermediate Range Monitors				
a. Detector not full in	2, 5[k]	6	TRSR 3.3.2.1.2	NA
b. Upscale	2, 5[k]	6	TRSR 3.3.2.1.1 TRSR 3.3.2.1.2 TRSR 3.3.2.1.5	$\leq$ 110/125 divisions of full scale
c. Inop	2, 5[k]	6	TRSR 3.3.2.1.2	NA
d. Downscale[f]	2, 5[k]	6	TRSR 3.3.2.1.1 TRSR 3.3.2.1.2 TRSR 3.3.2.1.5	$\geq$ 3/125 divisions of full scale
3. Average Power Range Monitors				
a. Simulated Thermal Power - Upscale	1	3	TRSR 3.3.2.1.4 TRSR 3.3.2.1.8	0.62
1. Flow Biased				$\leq 0.63(W - \Delta W)$[g] + 58.5% 57.4%
2. High Flow Clamped				with a maximum of 110% RTP
b. Inop	1, 2	3	TRSR 3.3.2.1.4	NA
c. Neutron Flux - Downscale	1	3	TRSR 3.3.2.1.4 TRSR 3.3.2.1.8	$\geq$ 3% RTP
d. Simulated Thermal Power - Upscale (Setdown)	2	3	TRSR 3.3.2.1.4 TRSR 3.3.2.1.8	$\leq$ 14% RTP
e. Flow - Upscale	1	3	TRSR 3.3.2.1.4 TRSR 3.3.2.1.8	$\leq$ 113% rated flow

(continued)

(f) This Function shall be automatically bypassed when the IRM channels are on range 1.

(g) The APRM Simulated Thermal Power - Upscale Flow Biased Rod Block setpoint varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW = 0% for two loop operation. ΔW = 8% for single loop operation.

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

Revision XX

TABLE TR3.3.2.1-2 (Page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	TRIP SETPOINT
1. Source Range Monitors	
a. Detector not full in	NA
b. Upscale	$\leq 1.0 \times 10^5$ cps
c. Inop	NA
d. Downscale	$\geq$ 3 cps[b]
2. Intermediate Range Monitors	
a. Detector not full in	NA
b. Upscale	$\leq$ 108/125 divisions of full scale
c. Inop	NA
d. Downscale	$\geq$ 5/125 divisions of full scale
3. Average Power Range Monitor	0.62 54.5%
a. Simulated Thermal Power - Upscale	
1) Flow Biased	$\leq 0.63(W - \Delta W)$[a] + 55.6%,
2) High Flow Clamped	with a maximum of 108% RTP
b. Inop	NA
c. Neutron Flux - Downscale	$\geq$ 5% RTP
d. Simulated Thermal Power - Upscale (Setdown)	$\leq$ 12% RTP
e. Flow - Upscale	$\leq$ 110% rated flow
4. Scram Discharge Volume	
a. Water Level - High	$\leq$ 589 ft. 11 ½ inches
b. Scram Trip Bypass	NA

(a) The APRM Simulated Thermal Power - Upscale Flow Biased Rod Block setpoint varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW = 0% for two loop operation. ΔW = 8% for single loop operation.

(b) May be reduced to $\geq$ 0.7 cps provided the signal to noise ratio $\geq$ 20.

Revision XX

TR 3.3 INSTRUMENTATION

TR 3.3.7.3 Feedwater Flow Instrumentation

TRLCO 3.3.7.3 The Leading Edge Flow Meter instrumentation system shall be OPERABLE.

APPLICABILITY: MODE 1, with THERMAL POWER > 3430 MWt

ACTIONS

---------------------------------------NOTE---------------------------------------
TRLCO 3.0.4.b is not applicable for the Leading Edge Flow Meter
----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more systems inoperable.	A.1 Restore required instruments to OPERABLE status.	72 hours
B. REQUIRED ACTION and associated COMPLETION TIME of CONDITION A not met.	B.1 Reduce power to ≤ 3430 MWt.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TRSR 3.3.7.3.1 Perform CHANNEL CHECK.	12 hours

TR B3.3 INSTRUMENTATION

TR B3.3.7.3 Feedwater Flow Instrumentation

BASES

The highly accurate Leading Edge Flow Meter CheckPlus Instrumentation allowed an increase in Licensed Thermal Power from 3430 MWt to 3486 MWt by reducing instrument uncertainty. When one or both channels of this instrumentation is out of service, operation at 3486 MWt is allowed for up to 72 hours following discovery of an INOPERABLE channel. If the instrumentation cannot be repaired within 72 hours, then power must be reduced to and maintained no higher than 3430 MWt until the instrumentation is repaired. If a decrease in power to below 3430 MWt occurs during the 72 hour period, then power must be maintained no higher than 3430 MWt until the instrumentation is repaired.

Technical Specifications Bases Markup

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

References 14, 15, and 16 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate based algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

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

The OPRM Upscale Function is required to be OPERABLE when the plant is at ≥ ~~25%~~ 27.5% RTP, the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. Within this region, the automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 28% RTP and recirculation drive flow is < 60% of rated flow, the operating region where actual thermal-hydraulic oscillations may occur. The lower bound, 25% RTP, is chosen to provide margin in the unlikely event of a power increase transient that could occur without operator action while the plant is operating below the ~~28%~~ 27.5% automatic OPRM Upscale trip enable point.

An OPRM Upscale trip function trip is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with the period confirmations and relative cell amplitude exceeding specified setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel trip. An OPRM Upscale trip is also issued from the channel if either the growth rate or amplitude based algorithms detect growing oscillatory changes in the neutron flux for one or more cells in that channel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

signal from a level switch and a level transmitter to each RPS logic channel. The level measurement instrumentation satisfies the recommendations of Reference 8.

The Allowable Value is chosen low enough to ensure that there is sufficient volume in the SDV to accommodate the water from a full scram.

Four channels of each type of Scram Discharge Volume Water Level—High Function, with two channels of each type in each trip system, arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from these Functions on a valid signal. These Functions are required in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

9. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

Turbine Stop Valve—Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER $\geq$ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first

29.5%

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BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

stage pressure of ≥ 161.9 psig; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 30% RTP. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If the bypass channel is placed in the nonbypass condition, the Turbine Stop Valve-Closure Function is considered OPERABLE. 29.5%

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

Eight channels of Turbine Stop Valve-Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure-High and the Average Power Range Monitor Neutron Flux-Upscale Functions are adequate to maintain the necessary safety margins. 29.5% 29.5%

10. <u>Turbine Control Valve Fast Closure</u>

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

Turbine Control Valve Fast Closure signals are initiated by the de-energization of the solenoid dump valve at each control valve. Redundant relay signals are provided to each RPS logic channel such that fast closure of one control valve in each RPS trip system will initiate a scram. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure of 29.5%
≥ 161.9 psig; therefore, to consider this Function

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BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY (continued)

OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 30% RTP. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If the bypass channel is placed in the nonbypass condition, the Turbine Control Valve Fast Closure Function is considered OPERABLE.

29.5%

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BASES

SURVEILLANCE REQUIREMENTS (continued)

The 18 month Frequency of SR 3.3.1.1.15 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

Additionally, the 24 month Frequency of SR 3.3.1.1.19 is based on Reference 13.

SR 3.3.1.1.16 29.5%

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Additionally, consideration is given to the fact that main turbine bypass flow can affect this setpoint 29.5% nonconservatively (THERMAL POWER is derived from turbine first stage pressure; where turbine first stage pressure of 161.9 psig conservatively correlates to 30% RTP), the main turbine bypass valves must remain closed at THERMAL POWER ≥ 30% RTP to ensure that the calibration remains valid.

29.5% 29.5%

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 30% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment, reliability of the components, and ≥ 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.18

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For the APRM Simulated Thermal Power - Upscale Function, this SR also includes calibrating the associated recirculation loop flow channel.

SR 3.3.1.1.18 is modified by a Note that states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.3) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8).

The Frequency of SR 3.3.1.1.18 is based upon 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

Insert 1 Here

SR 3.3.1.1.20

This SR ensures that scrams initiated from the OPRM Upscale Function (Function 2.f) will not be inadvertently bypassed when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ ~~28%~~ 27.5% RTP and recirculation drive flow is < 60% rated flow. This normally involves confirming the bypass setpoints. The bypass setpoint values are considered to be nominal values as discussed in Reference 20, and have been adjusted for power uprate. The surveillance ensures that the OPRM Upscale Function is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow.

If any bypass setpoint is nonconservative (i.e., the OPRM Upscale Function is bypassed when APRM Simulated Thermal Power ≥ ~~28%~~ 27.5% and recirculation drive flow < 60% rated), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the bypass setpoint may be adjusted to place the channel in a conservative condition (unbypassed). If placed in the unbypassed condition, this

Insert 1

Surveillance Requirement SR 3.3.1.1.18 for Function 2.b is modified by two Notes as identified in Table 3.3.1.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be within the as-left tolerance of the Nominal Trip Setpoint (NTSP). Where a setpoint more conservative than the NTSP is used in the plant surveillance procedures (field setting), the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as-left tolerance of the NTSP, then the channel shall be declared inoperable. The second Note also requires that the NTSPs and the methodologies for calculating the as-left and the as-found tolerances be in the Technical Requirements Manual.

Enclosure 4
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Revised (Clean) Operating License and Technical Specifications Pages

8 Pages

(4) DECo, 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 such 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) DECo, 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) DECo, 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 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

DECo is authorized to operate the facility at reactor core power levels not in excess of 3486 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No.   and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DECo shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between Detroit Edison Company and

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, and 2.f. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 29.5% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.12	-----------------NOTE----------------- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --------------------------------------- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.14	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure Functions are not bypassed when THERMAL POWER is ≥ 29.5% RTP.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.17	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 5 "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. ---------- Verify the RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.1.1.18	-----NOTE----- Neutron detectors are excluded. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.19	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.20	Verify OPRM is not bypassed when APRM Simulated Thermal Power is ≥ 27.5% and recirculation drive flow is < 60% of rated recirculation drive flow.	24 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5[(a)]	3	I	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5[(a)]	3	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux - Upscale (Setdown)	2	3[(c)]	G	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. Simulated Thermal Power - Upscale	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18(d)(e)	≤ 0.62 (W-ΔW) + 63.1% RTP and ≤ 115.5% RTP[(b)]

(continued)

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

(b) ΔW = 8% when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." Otherwise ΔW = 0%.

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

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

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

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Scram Discharge Volume Water Level - High					
a. Level Transmitter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft, 0 inches
	5[a]	2	I	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft, 0 inches
b. Float Switch	1,2	2	G	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft, 0 inches
	5[a]	2	I	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 596 ft, 0 inches
9. Turbine Stop Valve - Closure	≥ 29.5% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	≤ 7% closed
10. Turbine Control Valve Fast Closure	≥ 29.5% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17	NA
11. Reactor Mode Switch - Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.15	NA
	5[a]	2	I	SR 3.3.1.1.13 SR 3.3.1.1.15	NA
12. Manual Scram	1,2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5[a]	2	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched recirculation loop jet pump flows shall be in operation;

OR

One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;
3. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power–Upscale) Allowable Value of Table 3.3.1.1-1 is reset for single loop operation, when in MODE 1; and
4. THERMAL POWER is ≤ 66.1% RTP.

---------------------------NOTE---------------------------
Application of the required limitations for single loop operation may be delayed for up to 4 hours after transition from two recirculation loop operations to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

Enclosure 5
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Regulatory Issue Summary (RIS) 2002-03 Cross Reference

18 Pages

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty

I.1	Detailed description of plant-specific implementation of feedwater flow measurement technique and power increase gained as a result of implementing technique	Enclosure 1	3.1 3.2	Background and General Approach LEFM Flow Measurement and Core Thermal Power Uncertainty
I.1.A	NRC approval of topical report on flow measurement technique	Enclosure 1	3.2.1	LEFM Flow Measurement
I.1.B	Reference to NRC's approval of proposed measurement technique	Enclosure 1	3.2.1	LEFM Flow Measurement
I.1.C	Plant Implementation	Enclosure 1	3.2.2	Plant Implementation
I.1.D	Disposition of NRC criteria	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
I.1.E	Total power measurement uncertainty calculation for the plant	Enclosure 1 Enclosure 13	3.2.3	LEFM and Core Thermal Power Measurement Uncertainty and Methodology Core Thermal Power Uncertainty Calculation
I.1.F	Calibration and maintenance	Enclosure 1	3.2.4 3.2.5	Disposition of NRC Criteria for Use of LEFM Topical Reports Deficiencies and Corrective Actions

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
I.1.G	Proposed outage time for LEFM and basis for selected time	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
		Enclosure 3		Markup of Proposed Technical Requirements Manual Pages
I.1.H	Proposed actions if outage time is exceeded, and basis for actions	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
		Enclosure 3		Markup of Proposed Technical Requirements Manual Pages

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		**Fermi 2 MUR LAR**		
Section	**Description**	**Document**	**Section**	**Title / Description**

II. Accidents and Transients For Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level

II.1	Matrix for bounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations

III. Accidents and Transients For Which the Existing Analyses of Record Do Not Bound Plant Operation at the Proposed Uprated Power Level

III.1	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations
III.2	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations
III.3	Matrix for unbounded accidents and transients	Enclosure 7	9.0	Reactor Safety Performance Evaluations

IV. Mechanical/Structural/Material Component Integrity and Design

IV.1.A.i	Reactor vessel, nozzles, supports	Enclosure 7	3.2 3.2.1 3.2.2	Reactor Vessel Fracture Toughness Reactor Vessel Structural Evaluation

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.ii	Reactor core support structures and vessel internals	Enclosure 7	3.3	Reactor Internals
			3.3.1	Reactor Internal Pressure Difference
			3.3.2	Reactor Internals Structural Evaluation
			3.3.3	Steam Separator and Dryer Performance
			3.4	Flow-Induced Vibration
		Enclosure 1	3.4.2	Adverse Flow Effects
IV.1.A.iii	Control rod drive mechanisms	Enclosure 7	2.5	Reactivity Control

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	Enclosure 7	3.4	Flow-Induced Vibration
			3.5	Piping Evaluation
			3.5.1	Reactor Coolant Pressure Boundary Piping
			3.6	Reactor Recirculation System
			3.7	Main Steam Line Flow Restrictors
			3.8	Main Steam Isolation Valves
			3.9	Reactor Core Isolation Cooling
			3.10	Residual Heat Removal System
			3.11	Reactor Water Cleanup System

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		**Fermi 2 MUR LAR**		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.v	Balance of plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, and containment systems)	Enclosure 7	3.5 3.5.2 6.4.1 6.4.3	Piping Evaluation Balance-of-Plant Piping Evaluation Service Water Systems Reactor / Safety Auxiliaries Closed Cooling Water System
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	N/A	N/A	N/A
IV.1.A.vii	Reactor coolant pumps	N/A	N/A	N/A
IV.1.A.viii	Pressurizer shell, nozzles, and surge lines	N/A	N/A	N/A
IV.1.A.ix	Safety-related valves	Enclosure 7	3.1 3.8 4.1 4.1.1 4.1.2 6.5	Nuclear System Pressure Relief / Overpressure Protection Main Steam Isolation Valves Containment System Performance Generic Letter 89-10 Program Generic Letter 95-07 Program Standby Liquid Control System

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.i	Stresses	Enclosure 7	3.2	Reactor Vessel
			3.2.2	Reactor Vessel Structural Evaluation
			3.4	Flow-Induced Vibration
			3.5	Piping Evaluation
			3.5.1	Reactor Coolant Pressure Boundary Piping
			3.5.2	Balance-of-Plant Piping Evaluation
IV.1.B.ii	Cumulative usage factors	Enclosure 7	3.2.2	Reactor Vessel Structural Evaluation
IV.1.B.iii	Flow induced vibration	Enclosure 7	3.4	Flow-Induced Vibration
		Enclosure 1	3.4.2	Adverse Flow Effects

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		**Fermi 2 MUR LAR**		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.iv	Changes in temperature (pre- and post-uprate)	Enclosure 7	1.3	TPO Plant Operating Conditions
			1.3.1	Reactor Heat Balance
			1.3.2	Reactor Performance Improvement Features
			Table 1-2	Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.v	Changes in pressure (pre- and post-uprate)	Enclosure 7	1.3	TPO Plant Operating Conditions
			1.3.1	Reactor Heat Balance
			1.3.2	Reactor Performance Improvement Features
			Table 1-2	Thermal-Hydraulic Parameters at TPO Uprate Conditions

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	Description	Document	Section	Title / Description
IV.1.B.vi	Changes in flow rate (pre- and post-uprate)	Enclosure 7	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vii	High-energy line break locations	Enclosure 7	10.1 10.1.1 10.1.2	High Energy Line Break Steam Line Breaks Liquid Line Breaks
IV.1.B.viii	Jet impingement and thrust forces	Enclosure 7	10.1 10.1.1 10.1.2 10.1.2.7	High Energy Line Break Steam Line Breaks Liquid Line Breaks Pipe Whip and Jet Impingement
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	Enclosure 7	3.1	Nuclear System Pressure Relief / Overpressure Protection

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	Description	Document	Section	Title / Description
IV.1.C.ii	Reactor vessel fluence evaluation	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure temperature limit curves	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.iv	Reactor vessel low temperature overpressure protection	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.v	Reactor vessel upper shelf energy	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	Enclosure 7	3.2 3.2.1	Reactor Vessel Fracture Toughness

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.D	Code of record	Enclosure 7	3.2 3.2.2 3.5 3.5.1 3.5.2	Reactor Vessel Reactor Vessel Structural Evaluation Piping Evaluation Reactor Coolant Pressure Boundary Piping Balance-of-Plant Piping Evaluation
IV.1.E	Component inspection / testing programs and erosion / corrosion programs	Enclosure 7	3.5 3.5.1 3.5.2 10.6	Piping Evaluation Reactor Coolant Pressure Boundary Piping Balance-of-Plant Piping Evaluation Plant Life
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	N/A	N/A	N/A

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

V. Electrical Equipment Design

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

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	Description	Document	Section	Title / Description

VI. System Design

Section	Description	Document	Section	Title / Description
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling)	Enclosure 7	3.4	Flow-Induced Vibration
			3.5	Piping Evaluation
			3.5.1	Reactor Coolant Pressure Boundary Piping
			3.5.2	Balance-of-Plant Piping Evaluation
			3.6	Reactor Recirculation System
			3.7	Main Steam Line Flow Restrictors
			3.8	Main Steam Isolation Valves
			3.9	Reactor Core Isolation Cooling
			3.10	Residual Heat removal System
			3.11	Reactor Water Cleanup System

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
V.1.B	Containment systems	Enclosure 7	4.1 4.1.1 4.1.2 4.1.3	Containment System Performance Generic Letter 89-10 Program Generic Letter 95-07 Program Generic Letter 96-06
V.1.C	Safety-related cooling water systems	Enclosure 7	6.4 6.4.1 6.4.5	Water Systems Service Water Systems Ultimate Heat Sink
V.1.D	Spent fuel pool storage and cooling systems	Enclosure 7	6.3 6.3.1 6.3.2 6.3.3 6.3.4	Fuel Pool Fuel Pool Cooling Crud Activity and Corrosion Products Radiation Levels Fuel Racks

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
V.1.E	Radioactive waste systems	Enclosure 7	4.5	Standby Gas Treatment System
			8.1	Liquid and Solid Waste Management
			8.2	Gaseous Waste Management
			8.3	Radiation Sources in the Reactor Core
			8.4	Radiation Sources in Reactor Coolant
			8.4.1	Coolant Activation Products
			8.4.2	Activated Corrosion Products
			8.4.3	Fission Products
			8.5	Radiation Levels
			8.6	Normal Operation Off-Site Doses
V.1.F	Engineered safety features (ESFs) heating, ventilation, and air conditioning systems	Enclosure 7	4.4	Main Control Room Atmosphere Control System
			6.6	Power Dependent Heating, Ventilation and Air Conditioning

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		**Fermi 2 MUR LAR**		
Section	**Description**	**Document**	**Section**	**Title / Description**

VII. Other

VII.1	Operator actions and effects on time available	Enclosure 7	4.1 6.7 9.3 9.3.2 10.5	Containment System Performance Fire Protection Special Events Station Blackout Operator Training and Human Factors
VII.2.A	Emergency and abnormal operating procedures	Enclosure 7	10.9	Emergency Operating Procedures
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	Enclosure 1 Enclosure 7	3.2.4 3.4.3 10.5	Disposition of NRC Criteria for Use of LEFM Topical Reports Plant Modifications Operator Training and Human Factors
VII.2.C	The control room plant reference simulator	Enclosure 7	10.5	Operator Training and Human Factors
VII.2.D	The operator training program	Enclosure 7	10.5	Operator Training and Human Factors
VII.3	Modifications completion	Enclosure 1 Enclosure 6	3.4.3	Plant Modifications Summary of Regulatory Commitments
VII.4	Procedure Revisions – Licensed Power Level	Enclosure 1	3.2.6	Reactor Power Monitoring

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		Fermi 2 MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
VII.5.A	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on types and amounts of effluents released offsite, and whether bounded by final environmental statement and previous Environmental Assessments for the plant	Enclosure 1 Enclosure 7	5.0 6.4.2.1 8.6	Environmental Consideration Discharge Limits Normal Operation Off-Site Doses
VII.5.B	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on individual and cumulative occupational radiation exposure	Enclosure 1 Enclosure 7	5.0 8.5	Environmental Consideration Radiation Levels

VIII. Changes To Technical Specifications, Protection System Settings, And Emergency System Settings

VIII.1	A detailed discussion of each change to the plant's Technical Specifications, protection system settings, and/or emergency system settings needed to support the power uprate	Enclosure 1 Enclosure 2	1.0 2.0	Description Proposed Changes Markup of Proposed Operating License and Technical Specifications Pages

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

NRC REQUIREMENT		FERMI 2 RESPONSE		
NRC RIS 2002-03		**Fermi 2 MUR LAR**		
Section	**Description**	**Document**	**Section**	**Title / Description**
VIII.1.A	Description of the change	Enclosure 1	1.0	Description
			2.0	Proposed Changes
		Enclosure 2		Markup of Proposed Operating License and Technical Specifications Pages
VIII.1.B	Identification of analyses affected by and/or supporting the change	Enclosure 1	3.3	Evaluation of Changes to Operating License and Technical Specifications
		Enclosure 7		GEH Nuclear Energy Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization, NEDC-33578P
VIII.1.C	Justification for the change, including the type of information discussed in Section III, above, for any analyses that support and/or are affected by change	Enclosure 1	3.3	Evaluation of Changes to Operating License and Technical Specifications
		Enclosure 7		GEH Nuclear Energy Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization, NEDC-33578P

Enclosure 6
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Summary of Regulatory Commitments

2 Pages

Summary of Regulatory Commitments

The following table identifies commitments made by DTE Electric Company (DTE) in this document. Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.

COMMITMENT		COMMITTED DATE OR OUTAGE	ONE-TIME ACTION (Yes/No)	ON-GOING COMMITMENT (Yes/No)
1	Limitations regarding operation with an inoperable LEFM system will be included in the TRM. (Enclosure 1, Section 3.2.4)	Prior to Startup from the Sixteenth Refueling Outage	No	Yes
2	A process will be implemented to use the LEFM feedwater flow to adjust or correct the existing feedwater flow venturi-based signals. (Enclosure 1, Section 3.2.4)	Prior to Startup from the Sixteenth Refueling Outage	No	Yes
3	Plant maintenance and calibration procedures will be revised to incorporate Cameron’s maintenance and calibration requirements. Initial preventive maintenance scope and frequency will be based on vendor recommendations. (Enclosure 1, Section 3.2.4)	Prior to Startup from the Sixteenth Refueling Outage	No	Yes
4	Modifications for the power uprate will be implemented. (Enclosure 1, Section 3.4.3)	Prior to Startup from the Sixteenth Refueling Outage	Yes	No

Summary of Regulatory Commitments

COMMITMENT		COMMITTED DATE OR OUTAGE	ONE-TIME ACTION (Yes/No)	ON-GOING COMMITMENT (Yes/No)
5	Necessary procedure revisions for the power uprate will be completed. (Enclosure 1, Sections 3.4.4 and 3.4.6)	Prior to Startup from the Sixteenth Refueling Outage	No	Yes
6	The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes. (Enclosure 1, Section 3.4.6)	Prior to Startup from the Sixteenth Refueling Outage	Yes	No
7	Operator training will be completed prior to implementation of the proposed power uprate changes. (Enclosure 1, Section 3.4.6)	Prior to Startup from the Sixteenth Refueling Outage	Yes	No
8	Plant testing for the proposed changes will be completed as described in Enclosure 7, Section 10.4, "Testing." (Enclosure 1, Section 3.4.7)	As described	Yes	No
9	Plant-specific analyses for all potentially limiting events will be performed on a cycle-specific basis as part of the reload licensing process. (Enclosure 7, Section 2.2)	Prior to Startup from the Sixteenth Refueling Outage	Yes	No

Enclosure 8
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Affidavits from General Electric-Hitachi Nuclear Energy and the Electric Power Research Institute

Affidavit from General Electric-Hitachi (GEH) Nuclear Energy

Electric Power Research Institute (EPRI) Request for Withholding of the Proprietary Information Included in NEDC-33578P, Revision 0

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

NEIL WILMSHURST
Vice President and
Chief Nuclear Officer

February 1, 2013

Document Control Desk
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Request for Withholding of the following Proprietary Information Included in:

Safety Analysis Report for Fermi Generating Station UNIT 2 Thermal Power Optimization

NEDC-33578P
Revision 0
DRF Section 0000-0115-4329 R2
December 2012

To Whom It May Concern:

This is a request under 10 C.F.R. §2.390(a)(4) that the U.S. Nuclear Regulatory Commission ("NRC") withhold from public disclosure the report identified in the enclosed Affidavit consisting of the proprietary information owned by Electric Power Research Institute, Inc. ("EPRI") identified in the attached report. Proprietary and non-proprietary versions of the Report and the Affidavit in support of this request are enclosed.

EPRI desires to disclose the Report in confidence to assist the NRC review of the enclosed submittal to the NRC by DTE Energy. The Report is not to be divulged to anyone outside of the NRC or to any of its contractors, nor shall any copies be made of the Report provided herein. EPRI welcomes any discussions and/or questions relating to the information enclosed.

If you have any questions about the legal aspects of this request for withholding, please do not hesitate to contact me at (704) 704-595-2732. Questions on the content of the Report should be directed to Andy McGehee of EPRI at (704) 502-6440.

Sincerely,

1300 West W.T. Harris Boulevard, Charlotte, NC 28262-8550 USA • 704.595.2732 • Mobile 704.490.2653 • nwilmshurst@epri.com

ELECTRIC POWER RESEARCH INSTITUTE

AFFIDAVIT

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

Safety Analysis Report for Fermi Generating Station UNIT 2 Thermal Power Optimization
NEDC-33578P
Revision 0
DRF Section 0000-0115-4329 R2
December 2012

I, Neil Wilmshurst, being duly sworn, depose and state as follows:

I am the Vice President and Chief Nuclear Officer at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, California ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI requests that the Report be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information:

a. The Report is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Report do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Report. The Report is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Report contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information. EPRI made a substantial economic investment to develop the Report and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Report. If the Report were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Report for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Report.

c. EPRI's classification of the Report as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

> "'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:

(1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and

(2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

d. The Report contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Report was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Report. EPRI was required to devote these resources and effort to derive the Report. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Report is highly valuable to EPRI.

e. A public disclosure of the Report would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Report both domestically and internationally. The Report can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 1300 W WT Harris Blvd being the premises and place of business of Electric Power Research Institute, Inc.

Date: 2-1-2013.

Neil Wilmshurst

(State of North Carolina)
(County of Mecklenburg)

Subscribed and sworn to (or affirmed) before me on this 1st day of February, 2013, by Neil Wilmshurst, proved to me on the basis of satisfactory evidence to be the person(s) who appeared before me.

Signature Deborah H. Rouse (Seal)

My Commission Expires 2nd day of April, 2016.

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Jerald G. Head, state as follows:

(1) I am the General Manager of Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report, NEDC-33578P, "Safety Analysis Report for Fermi Generating Station Unit 2 Thermal Power Optimization," Revision 0, January 2013. The GEH proprietary information in NEDC-33758P is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and large equation objects containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains the detailed GEH methodology for thermal power optimization for GEH Boiling Water Reactors (BWRs). Development of these methods, techniques, and information and their application for the design, modification, and analyses methodologies and processes was achieved at a significant cost to GEH.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience databases that constitute major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 1st day of February 2013.

Jerald Head

Jerald G. Head
General Manager, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd.
Wilmington, NC 28401
Jerald.Head@ge.com

Enclosure 12
to NRC-13-0004

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Affidavits from Cameron International Corporation

Application for Withholding Proprietary Information from Public Disclosure for Caldon® Ultrasonics Engineering Report ER-781, Revision 2 (CAW 12-10)

Application for Withholding Proprietary Information from Public Disclosure for Caldon® Ultrasonics Engineering Report ER-818, Revision 0 (CAW 12-05)

Measurement Systems

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

CAMERON

December 13, 2012
CAW 12-10

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon® Ultrasonics Engineering Report ER-781 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Fermi Unit 2 Using the LEFM CheckPlus C System"

Gentlemen:

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

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 12-10 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 12-10 and should be addressed to the undersigned.

Very truly yours,

Ernest Hauser
Director of Sales

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

December 13, 2012
CAW 12-10

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Ernest Hauser
Director of Sales

Sworn to and subscribed before me

this 13th day of

December, 2012

Joann B. Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2015
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

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

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

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

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Cameron's competitors without license from Cameron constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Cameron, its customer or suppliers.

(e) It reveals aspects of past, present or future Cameron or customer funded development plans and programs of potential customer value to Cameron.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Cameron system, which include the following:

(a) The use of such information by Cameron gives Cameron a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Cameron competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Cameron ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Cameron at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Cameron of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Cameron in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Cameron capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10 CFR §§ 2. 390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld are the submittals titled:

- Caldon® Ultrasonics Engineering Report ER-781 Rev. 2 "Bounding Uncertainty Analysis for Thermal Power Determination at Fermi Unit 2 Using the LEFM CheckPlus C System"

It is designated therein in accordance with 10 CFR §§ 2.390(b)(1)(i)(A,B), with the reason(s) for confidential treatment noted in the submittal and further described in this affidavit. This information is voluntarily submitted for use by the NRC Staff in their review of the accuracy assessment of the proposed methodology for the LEFM CheckPlus C System used by Fermi Unit 2 for MUR UPRATES.

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

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

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

Further the deponent sayeth not.

Measurement Systems

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

CAMERON

June 13, 2012
CAW 12-05

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

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon® Ultrasonics Engineering Report ER-818 Rev. 0 "Meter Factory Calculation and Accuracy Assessment for Fermi Unit 2"

Gentlemen:

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

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.390, Affidavit CAW 12-05 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Cameron, be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW 12-05 and should be addressed to the undersigned.

Very truly yours,

Ernest Hauser
Director of Sales

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

June 13, 2012
CAW 12-05

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

Ernest Hauser
Director of Sales

Sworn to and subscribed before me

this 13th day of

June, 2012

Joann B. Thomas
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Joann B. Thomas, Notary Public
Findlay Twp., Allegheny County
My Commission Expires July 28, 2015
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

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

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

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

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

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

   (ii) The information is of a type customarily held in confidence by Cameron and not customarily disclosed to the public. Cameron has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Cameron policy and provides the rational basis required. Furthermore, the information is submitted voluntarily and need not rely on the evaluation of any rational basis.

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

Further the deponent sayeth not.

Enclosure 15
to NRC-13-0004

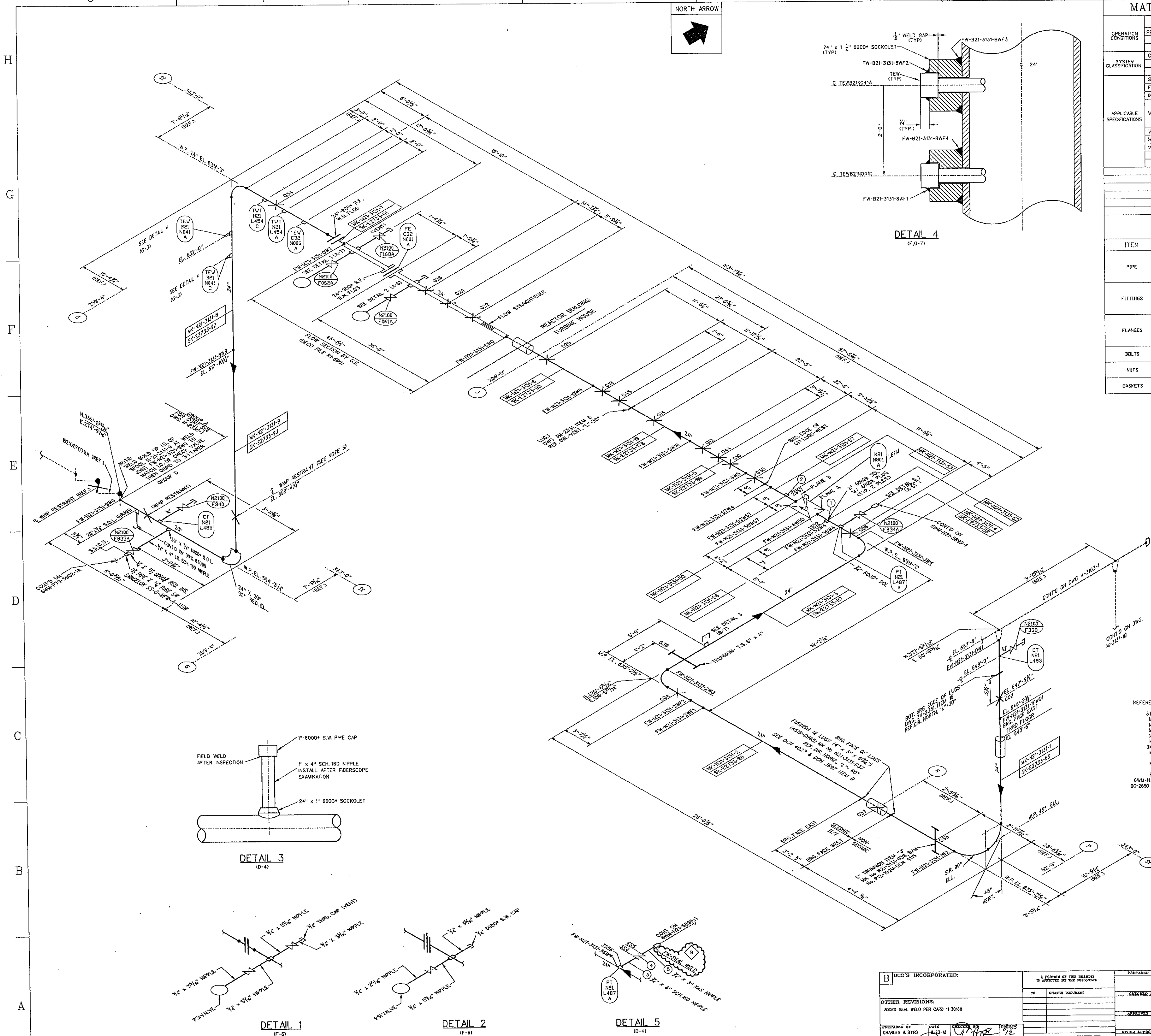
Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate

Drawings Describing the Installation of the LEFM

2 Drawings

Drawing 6M721-3131-1A, Revision B
Drawing 6M721-3131-1B, Revision A



MATERIAL SPECIFICATIONS & NOTES

	SYSTEM	PRESSURE (PSIG)	TEMPERATURE (°F)
OPERATION CONDITIONS	FEEDWATER	DESIGN: 1275 NORMAL: 1116	DESIGN: 450 NORMAL: 425
		DESIGN: NORMAL:	DESIGN: NORMAL:
		DESIGN: NORMAL:	DESIGN: NORMAL:
SYSTEM CLASSIFICATION	CODE: ANSI B31.1.0	GROUP: D SEISMIC: NONE	QA: NO
		GROUP: D SEISMIC: II/I	QA: NO
APPLICABLE SPECIFICATIONS	SYSTEM DIAGRAMS: M-2023	DESIGN:	
	FABRICATION: DE SPEC 3071-030	ERECTION: DE SPEC 3071-031	
	INSULATION: DE SPEC 3071-032, CLASS 6		
	WELD END PREP	■ BUTT WELD ■ SOCKET WELD ☐ THREADED	☐ BACKING RING ■ INERT ARC WELD ☐ / ■ D.E. STD P439 IGAW ☐ ANSI B16 25 ☐ SEE NOTE 1
	WELDING PER FERMI WELDING MANUAL AND DE SPEC 312		
	HANGERS		
	INSTRUMENTATION		

MATERIAL

ITEM	SIZE	RATING	TYPE	ASTM/ANSI SPEC
PIPE	2" & SMALLER	SCH 160	SEAMLESS	A106, GR B
	2 1/2" THRU 24"	SCH 80	SEAMLESS	A106, GR B
	3/4" (INSTR)	XXS	SEAMLESS	A106, GR B
FITTINGS	2" & SMALLER	6000 LB	SW	A105
	2 1/2" THRU 24"	SCH 80	BW	A234 GR WPB-
FLANGES	2" & SMALLER	1500 LB	SW RF	A105
	2 1/2" THRU 24"	900 LB	WN RF	A105
BOLTS	ALL		BOLT STUDS	A193 GR B7
NUTS	ALL		HEX	A193 GR 2H
GASKETS	2" & SMALLER		FLEXITALLIC	STYLE C-G
	2 1/2" & LARGER		FLEXITALLIC	STYLE C-G

INSTRUMENTATION:
1. INSTRUMENTATION AS SHOWN AND PER CECO.

NOTES:
1. ARROWS INDICATE THE DIRECTION OF THE FLOW.
2. BUTTWELDED JOINTS SHALL BE PROPERLY BEVELED AND INTERNALLY MACHINED IN ACCORDANCE WITH EDISON STD. P439 FOR INERT GAS ARC WELDING.
3. ALL PIPE SPOOLS, WELDS, ETC., SHALL BE MARKED WITH THE FOLLOWING PLANT IDENTIFICATION
   SYSTEM:N21, DWG. No:3131, IDENT No 1, 2, 3, ETC.
4. FOR SUPPORT SYSTEM COMPONENT STRUCTURAL CALCULATIONS SEE DESIGN CALCULATION NUMBER DC-0673.
5. PIPE WHIP RESTRAINTS SHOWN ON THIS LINE ARE SHOWN ON DRAWINGS C-2538 & C-2542.
6. ALL SUPPORT NUMBERS ARE PREFIXED BY "N21-3131-"

REFERENCE DRAWINGS:

3T-N21-3131	LARGE BORE TRIM INSTALLATION INSTRUCTIONS
M-2320	KEY PLAN OF PIPING SECTION AND PLANS
M-2054	REACTOR BLDG-PIPING PLAN-FIRST FLOOR
M-3014	TURBINE HSE-SECTION-ROW 11
M-2378	TURBINE HSE-PIPING PLAN-2ND FLOOR-COL 12-17
M-3103	FEEDWATER NORTH SOUTH HTR 6 DISCH VALVE
3M-2331	STD PIPE LUGS FOR HGRS
M-3131-1B	ISOMETRIC-FEEDWATER FROM 36" HEADER TO RX ISOLATION CK VLV B2100F076B
M-5715-1	REACTOR FEEDWATER SYSTEM FUNCTIONAL OPERATING SKETCH
R1-660	OUTLINE AND ASSEMBLY FOR FLOW TUBE SECTIONS
6WM-N21-5899-1	FEEDWATER SUPPLY TO PRESSURE TRANSMITTERS N21N802A/B
DC-2660 VOL IA DCD 1	PIPE STRESS CALCULATION

6M721-3131-1A
LATEST REVISION B

THIS DRAWING REPLACES: M-3131-1, REV. T & M-3131-2, REV. J

NON-NUCLEAR SAFETY RELATED

THIS IS A MICROSTATION PRODUCED DRAWING. CHANGES OR REVISIONS MUST BE BROUGHT TO THE ATTENTION OF THE PLANT ENGINEERING DESIGN GROUP TO ENSURE THAT CONFIGURATION CONTROL IS MAINTAINED.

B DCD'S INCORPORATED:

OTHER REVISIONS:
ADDED SEAL WELD PER CARD 11-30168

PREPARED BY	DATE	CHECKED BY	DATE
CHARLES H. BYRD	8-23-12	[signature]	8-23-12
APPROVED BY			DATE
[signature]			8-23-2012

A PORTION OF THIS DRAWING IS AFFECTED BY THE FOLLOWING:

TC	CHANGE DOCUMENT

EDP POSTING ENTRIES REQUIRE THE ADDITION OF THE ARMS STATUS CODE: EITHER "AFC" OR "ASB"

PREPARED BY / DATE — CHECKED BY / DATE — APPROVED BY / DATE — OTHER APPROVALS / DATE

INC. CODE: S

Detroit Edison — Fermi 2

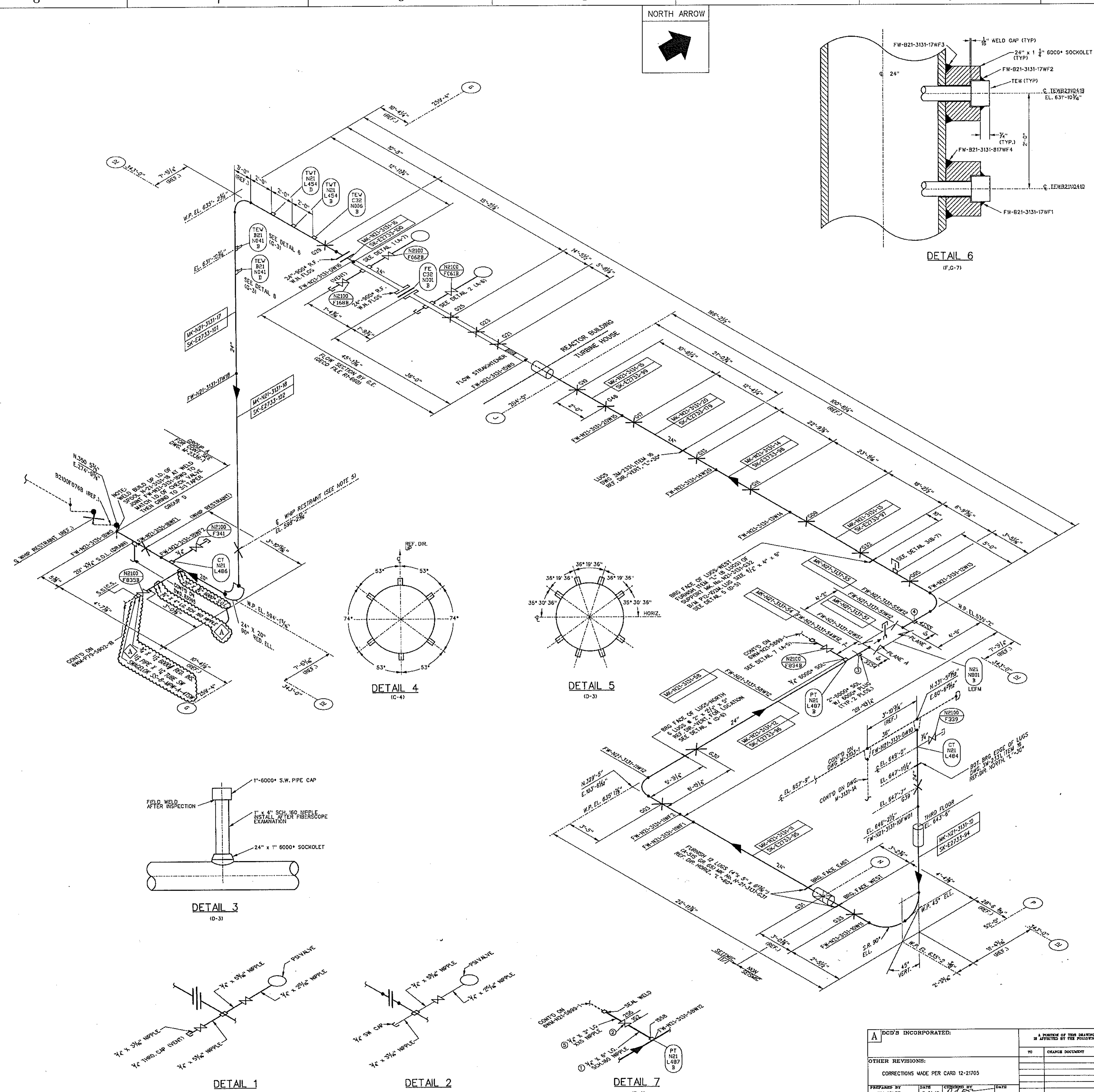
TITLE: PIPING ISOMETRIC–FEEDWATER FROM 36" MAIN HEADER TO REACTOR ISOLATION CHECK VALVE B2100F076A

APERTURE CARD TITLE: P/I FEEDWTR HDR RX ISO CHK VAL

PLANT IDENTIFICATION SYSTEM NUMBER: N2100 — ARMS RECIPIENT #

DOCUMENT TYPE CODE: DDDMEC — NUC OPS FILE NO. 1801 — DATE ISSUED TO IS: 8/24/12

DRAWING NUMBER: 6M721-3131-1A — REV. B



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6M721-3131-1B
LATEST REVISION A

THIS DRAWING REPLACED: M-3131-1, REV. T & M-3131-2, REV. J

NON-NUCLEAR SAFETY RELATED

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A DCD'S INCORPORATED:	
OTHER REVISIONS:	CORRECTIONS MADE PER CARD 12-21705
PREPARED BY: WJ ADLER	DATE: 3-21-12
CHECKED BY: DAVID KAKOCZKI	DATE: 3-27-12
APPROVED BY: JOHN MOSES FOR BILL MAYES PER TMPE 11-0012	DATE: 3-27-2012

A PORTION OF THIS DRAWING IS AFFECTED BY THE FOLLOWING:

TO	CHANGE DOCUMENT

EDP POSTING ENTRIES REQUIRE THE ADDITION OF THE ARMS STATUS CODE: EITHER "AFC" OR "ASB"

PREPARED BY	DATE
CHECKED BY	DATE
APPROVED BY	DATE
OTHER APPROVALS	DATE

INC.CODE	S
Detroit Edison	Fermi 2
TITLE	PIPING ISOMETRIC–FEEDWATER FROM 36" HEADER TO REACTOR ISOLATION CHECK VALVE B2100F076B
APERTURE CARD TITLE	P/I FEEDWTR HDR RX ISO CHK VAL
PLANT IDENTIFICATION SYSTEM NUMBER	N2100
ARMS RECIPIENT #	
DOCUMENT TYPE CODE	DDDMEC
NUC OPS FILE NO.	1801
DATE ISSUED TO IS	3-28-12
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REV.	A