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NRC-17-0012

10 CFR 50.90

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) General Electric Report NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," dated October 1992
 - 3) Letter from A. C. Thadani (U. S. Nuclear Regulatory Commission) to G. J. Beck (BWROG), "Acceptance for Referencing Topical Report NEDO-31400," dated May 15, 1991
 - 4) NRC Regulatory Guide 1.183 Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ML003716792)
 - 5) NRC Regulatory Guide 1.195 Revision 0, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," dated May 2003 (ML031490640)
 - 6) NRC Letter, "Fermi 2 - Issuance of Amendment Re: Reevaluation of Fuel Handling Accident, Selective Implementation of 10 CFR Part 50.67," dated September 28, 2001 (ML012290521)
 - 7) NRC Letter, "Fermi 2 - Issuance of Amendment Re: Selective Implementation of Alternative Radiological Source Term Methodology," dated September 28, 2004 (ML042430179)

Subject: License Amendment Request to Revise Technical Specifications to Eliminate Main Steam Line Radiation Monitor Reactor Trip and Primary Containment Isolation System Group 1 Isolation Functions

In accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," DTE Electric Company (DTE) requests amendment to Appendix A, Technical Specifications of Renewed Facility Operating License NPF-43 for Fermi Unit 2 (Fermi 2).

This submittal requests elimination of the main steam line radiation monitor (MSLRM) functions for initiating: 1) a reactor protection system (RPS) automatic reactor trip and 2) the associated (Group 1) primary containment isolation system (PCIS) automatic closure of the main steam isolation valves (MSIVs) and main steam line (MSL) drain valves. Specifically, the proposed Technical Specification (TS) changes remove requirements for the MSLRM trip function from TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation." The proposed changes also remove requirements for PCIS Group 1 isolation from TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation." However, the MSLRM isolation function in TS Table 3.3.6.1-1 is relocated and retained for the current existing PCIS Group 2 isolation of the reactor water sample line.

The justification for eliminating the MSLRM trip and isolation functions from initiating an automatic reactor trip and automatic closure of the MSIVs is based on the approach developed by the Boiling Water Reactor Owners Group (BWROG) as documented in General Electric (GE) Licensing Topical Report (LTR) NEDO-31400A (Reference 2). This LTR was reviewed and approved by the NRC (Reference 3) as acceptable for referencing by licensees requesting TS changes regarding elimination of the MSLRM trip and isolation functions.

The change framework established in NEDO-31400A involves re-evaluation of the control rod drop accident (CRDA) in Updated Final Safety Analysis Report (UFSAR) Section 15.4.9 assuming the MSIVs remain open considering two cases: 1) the original deterministic release from the main condenser at a rate of 1%/day and 2) forced release via the offgas system. Subsequent to the development of NEDO-31400A, the NRC issued updated guidance for determining the radiological consequences of the CRDA (e.g. References 4 and 5), which require the evaluation of additional forced release paths (e.g. turbine gland sealing steam exhaust). The scope of the CRDA re-analysis is, therefore, expanded to also comply with the current regulatory guidance. Specifically, the analysis is performed using the methods and assumptions of the Reference 4 Alternate Source Term to demonstrate compliance with 10 CFR 50.67 limits for onsite and offsite radiological consequences. Fermi 2 has previously selectively implemented the Alternate Source Term methodology for analysis of the design basis fuel handling and loss of coolant accidents (References 6 and 7). Accordingly, this submittal requests approval to extend the application of the Reference 4 Alternate Source Term assumptions and methods to the analysis of the Fermi 2 CRDA.

The elimination of the MSLRM isolation function from initiating an automatic closure of the MSL drain valves was not specifically discussed in NEDO-31400A; therefore, this submittal provides additional information to justify those specific TS changes.

This submittal also proposes the addition of two new TS Limiting Conditions for Operation (LCOs), 3.3.7.2 and 3.3.7.3, for the mechanical vacuum pump (MVP) and gland seal exhaustor (GSE) trip instrumentation that will be required to actuate in response to high MSL radiation. The results of the re-analysis of the CRDA established that these functions are needed to ensure that the onsite and offsite radiological consequences of a postulated CRDA remain with the regulatory acceptance criteria. As such, these trips satisfy the 10 CFR 50.36(c)(2)(ii) Criterion 3 requirements for inclusion in the plant TS.

The following additional changes support the implementation of the proposed elimination of the MSLRM reactor trip and PCIS Group 1 isolations:

1. Installation of a new trip of the GSEs on detection of high MSL radiation. This trip circuit will only be required to be operable in MODES 1 and 2, with GSEs in service, with any MSL not isolated, and reactor thermal power below 10%.
2. Reset of the Fermi 2 MSLRM and offgas 2-minute delay pipe radiation monitor alarm setpoints to 1.5 times the normal full power nitrogen-16 (N-16) background (with consideration of hydrogen addition and On-Line Noble Chemistry injection). Plant alarm response procedures will be revised as necessary to direct prompt sampling of the reactor coolant for contamination levels if the MSLRMs or offgas 2-minute delay pipe radiation monitors, or both, exceed their alarm setpoints, unless the condition is an expected response to known changes in plant operating conditions.
3. Reassignment of the MVP control logic, which currently trips these pumps automatically on detection of high radiation in the offgas system 2-minute delay pipe, to instead trip/isolate in response to detection of high MSL radiation.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan State Official.

Enclosure 1 provides a detailed description and evaluation of the proposed changes, including an analysis of the significant hazards considerations using the standards of 10 CFR 50.92. DTE has concluded that the changes proposed herein do not result in a significant hazards consideration. Enclosure 2 provides the existing TS pages marked up to show the proposed changes. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides a markup of the existing TS Bases pages. Changes to the existing TS Bases, consistent with the technical and regulatory analyses, will be implemented under the TS Bases Control Program. Enclosure 4 is provided for information only.

Enclosure 5 provides a compliance matrix against Regulatory Guide 1.183 for the re-analysis of the CRDA.

DTE requests approval of the proposed License Amendment by August 15, 2018, with the amendment being implemented during the next refueling outage following approval.

Should you have any questions or require additional information, please contact Mr. Scott A. Maglio, Manager – Nuclear Licensing at (734) 586-5076.

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

Executed on August 24, 2017



Keith J. Polson
Site Vice President

Enclosures:

1. Evaluation of the Proposed License Amendment
2. Marked-up Pages of Existing Fermi 2 TS
3. Clean Pages of Fermi 2 TS with Changes Incorporated
4. Marked-up Pages of Existing Fermi 2 TS Bases (For Information Only)
5. Compliance Matrix: Fermi 2 CRDA Re-Analysis vs. Regulatory Guide 1.183 Rev. 0

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Michigan Public Service Commission
Regulated Energy Division (kindschl@michigan.gov)

**Enclosure 1 to
NRC-17-0012**

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

**License Amendment Request to Revise Technical Specifications to
Eliminate Main Steam Line Radiation Monitor Reactor Trip and
Primary Containment Isolation System Group 1 Isolation Functions**

Evaluation of the Proposed License Amendment

Evaluation of the Proposed License Amendment

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Acronym List

AC	Alternating Current
AST	Alternate Source Term
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CRDA	Control Rod Drop Accident
DBA	Design Basis Accident
DDE	Deep Dose Equivalent
DTE	DTE Electric Company
EAB	Exclusion Area Boundary
EOP	Emergency Operating Procedure
FGR	Federal Guidance Report
GE	General Electric
GSE	Gland Seal Exhauster
GWD/MTU	GigaWatt-Days per Metric Ton of Uranium
LCO	Limiting Condition for Operation
LEFM	Leading Edge Flow Meter
LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
LTR	Licensing Topical Report
MCR	Main Control Room
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSLRM	Main Steam Line Radiation Monitor
MVP	Mechanical Vacuum Pump
NRC	Nuclear Regulatory Commission
OLNC	On-Line Noble Chemistry
PCIS	Primary Containment Isolation System
RG	Regulatory Guide
RPS	Reactor Protection System
RTP	Rated Thermal Power
SER	Safety Evaluation Report
SJAE	Steam-Jet Air Ejector
SR	Surveillance Requirement
SRP	Standard Review Plan
SSC	System, Structure, or Component
TEDE	Total Effective Dose Equivalent
TRM	Technical Requirements Manual
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 Code of Federal Regulations (CFR) 50.90, DTE Electric Company (DTE) is submitting a request for an amendment to the Technical Specifications (TS) for Fermi 2.

The proposed amendment would modify the Fermi 2 TS and associated TS Bases to eliminate the main steam line radiation monitor (MSLRM) functions for initiating: 1) a reactor protection system (RPS) automatic reactor trip and 2) the associated (Group 1) primary containment isolation system (PCIS) isolations, which includes automatic closure of the main steam isolation valves (MSIVs) and main steam line (MSL) drain valves. Specifically, the proposed TS changes remove the requirements for the MSLRM trip function from TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation." The proposed changes also remove requirements for PCIS Group 1 isolation from TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation." However, the MSLRM isolation function in TS Table 3.3.6.1-1 is relocated and retained for the current existing PCIS Group 2 isolation of the reactor water sample line.

This submittal also proposes the addition of two new TS limiting conditions for operation (LCOs), 3.3.7.2 and 3.3.7.3, for the mechanical vacuum pump (MVP) and gland seal exhaustor (GSE) trip instrumentation that will be required to actuate in response to high MSL radiation. In addition, this submittal includes an analysis of the control rod drop accident (CRDA) proposing to adopt the Regulatory Guide (RG) 1.183 Rev. 0 Alternate Source Term (AST) methodology.

2.0 BACKGROUND

Boiling Water Reactors (BWRs), including Fermi 2, are equipped with radiation monitors which are located near the MSLs downstream of the outboard MSIVs. The MSLRMs are designed to detect a release of fission products due to gross fuel failure. Conditions of high radiation in the MSLs are indicative of a design basis accident (DBA) CRDA. On detection of high radiation, the trip signals generated by the MSLRMs initiate a reactor scram, a PCIS Group 1 isolation of all MSIVs (four inboard/four outboard) and associated MSL drain line isolation valves, and a PCIS Group 2 isolation of the normally-closed inboard and outboard reactor water sample line isolation valves associated with the reactor recirculation loop B. The current reactor scram and PCIS nominal trip setpoint is 3.0 times normal full power nitrogen-16 (N-16) background, high enough above normal background radiation levels to prevent spurious trips, yet low enough to promptly detect gross failure in the fuel cladding. The current MSLRM alarm setpoint is approximately 2.8 times normal full power background radiation levels.

The nuclear industry experienced numerous inadvertent MSLRM-initiated reactor shutdowns from 1980 through October 1992. None of those reactor shutdowns were caused by fuel degradation; but were, instead, the result of instrument failures, chemistry excursions, radiation monitor maintenance errors, and other causes. In order to reduce the potential for unnecessary reactor trips and PCIS isolations caused by spurious actuation of the MSLRM trip and isolation functions, and to increase plant operational flexibility, the Boiling Water Reactor Owners Group

(BWROG) proposed to eliminate the RPS automatic reactor trip and MSIV closure functions initiated by the MSLRMs, and towards this end, provided a supporting safety analysis in NEDO-31400 (Reference 7.1). NEDO-31400 evaluated the role of the MSLRM in the CRDA, confirming that removal of the MSLRM trip and isolation functions would not compromise CRDA consequences. Two release paths were considered: the original Standard Review Plan (SRP) 15.4.9 condenser release (1% volume per day) and a forced release via the offgas system (i.e. steam-jet air ejector (SJAE) discharge through a series of sand filters and charcoal beds). NEDO-31400 demonstrated that the radiological consequences from either release path would be expected to satisfy 10 CFR 100 acceptance limits.

The Nuclear Regulatory Commission (NRC) safety evaluation report (SER) dated May 15, 1991 (Reference 7.2) accepted NEDO-31400 for use as a reference in licensee applications to eliminate the MSLRM reactor scram and MSIV closure, provided that:

1. Assumptions with regard to input values (including power per assembly, Chi/Q , and decay times) that are made in the generic [NEDO-31400 defined CRDA] analysis bound those for the specific plant.
2. There is sufficient evidence (such as implemented or proposed operating procedures or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines would be controlled expeditiously to limit both occupational doses and environmental releases.
3. The MSLRM and offgas radiation monitor alarm setpoints are set to 1.5 times the normal N-16 background dose rate at the monitoring locations, and commitments are made to promptly sample the reactor coolant for contamination levels if the MSLRM or offgas radiation monitors, or both, exceed their alarm setpoints.

Because not all BWRs have the MVP trip and MSL drain isolations on the high MSLRM radiation signal, NEDO-31400 did not address those specific features. In addition, the report did not include an analysis of the impact of eliminating the PCIS Group 1 isolation of the MSL drain valves.

The BWROG re-issued its guidance with the NRC SER attached as NEDO-31400A in October 1992 (Reference 7.1). Between March 1992 and October 1997, most domestic BWRs submitted a request for, and received NRC approval for, elimination of the MSLRM trip and isolation functions from initiating an automatic reactor scram and automatic closure of the MSIVs based on NEDO-31400A. Most recently, in December 2014, Peach Bottom Unit 3 experienced a half scram caused by a short that occurred during channel function testing of the MSLRM (Reference 7.4). Peach Bottom subsequently received approval to eliminate their MSLRM trip and isolation functions (Reference 7.5). DTE has elected to pursue the elimination of the MSLRM high radiation reactor scram and PCIS Group 1 isolations in order to eliminate this vulnerability to an inadvertent plant shutdown with loss of normal heat sink that might result from spurious failures of the MSLRM trip instrumentation.

3.0 DETAILED DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, DTE is submitting a request for an amendment to the TS for Fermi 2.

The proposed amendment would modify the Fermi 2 TS and associated TS Bases to eliminate the MSLRM from initiating: 1) a RPS automatic reactor trip (scram) and 2) the associated PCIS (Group 1) isolations, which include automatic closure of the MSIVs and MSL drain valves. Specifically, the proposed TS changes would:

1. Remove the requirement for the MSLRM trip function (i.e. Function 6) from TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation."
2. Remove Condition H from TS 3.3.1.1, "Reactor Protection System Instrumentation," since it was only referenced by the Function 6 item in Table 3.3.1.1-1 which is being removed as indicated above (#1).
3. Remove the MSLRM isolation function (i.e. Function 1.f) from TS Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation."
4. Relocate the MSLRM isolation function for PCIS Group 2 within TS Table 3.3.6.1-1 as new Function 2.d. The only change to the Function other than the relocation is to reference Condition F instead of Condition D.
5. Renumber the Manual Initiation function in TS Table 3.3.6.1-1 from 2.d to 2.e to support the relocation indicated above (#4).
6. Revise the Completion Time of Condition A for TS 3.3.6.1 to refer to Function 2.d instead of Function 1.f to support the relocation indicated above (#4).
7. Add new TS 3.3.7.2, "Mechanical Vacuum Pump (MVP) Trip Instrumentation."
8. Add new TS 3.3.7.3, "Gland Seal Exhauster (GSE) Trip Instrumentation."
9. Update the TS Table of Contents to reflect the new TS indicated above (#7 and #8).
10. Revise the associated TS Bases for the applicable TS and the corresponding Technical Requirements Manual (TRM) sections accordingly.

The MSLRM is designed to monitor radiation levels in the MSLs, since the presence of high radiation indicates a potential release of fission products due to gross fuel failure. Upon detection of high radiation, an alarm signal is initiated. Receipt of a "High-High" radiation signal currently initiates an RPS automatic reactor scram and a PCIS (Group 1) automatic closure of all MSIVs and MSL drain valves in order to limit fuel damage and contain the release of fission products. The "High-High" radiation signal also initiates a Group 2 PCIS isolation of the reactor water sample line. No changes are being proposed to eliminate this Group 2 PCIS isolation from the TS, only to relocate it within the TS. The MSLRM trip settings are selected high enough above full power background radiation levels to prevent spurious isolation and to increase the plant operational flexibility, yet low enough to promptly detect a gross release of fission products from the fuel.

The justification for eliminating the MSLRM trip and isolation functions from initiating an automatic reactor trip and automatic closure of the MSIVs is based on the approach developed by the BWROG as documented in General Electric (GE) Licensing Topical Report (LTR)

NEDO-31400A (Reference 7.1). This LTR was reviewed and approved by the NRC (Reference 7.2) as acceptable for referencing by licensees requesting TS changes regarding elimination of the MSLRM trip and isolation functions. The change framework established in NEDO-31400A requires evaluation of the CRDA in Updated Final Safety Analysis Report (UFSAR) Section 15.4.9 assuming the MSIVs remain open considering two cases: 1) the original deterministic release from the main condenser at a rate of 1%/day and 2) forced release via the offgas system. The general assumptions of the NEDO-31400A analysis follow from SRP 15.4.9. Subsequent to the development of NEDO-31400A, the NRC issued updated guidance for determining the radiological consequences of the CRDA using AST and non-AST methodologies (References 7.12 and 7.13). Fermi 2 has previously selectively implemented the AST methodology for analysis of the design basis fuel handling accident and loss of coolant accident (LOCA) (References 7.14 and 7.15). Accordingly, this submittal also requests approval to extend the application of the RG 1.183 AST assumptions and methods to the analysis of the Fermi 2 CRDA in UFSAR Section 15.4.9. The results of the re-analysis of the CRDA that has been prepared in support of this submittal will, therefore, demonstrate compliance with 10 CFR 50.67.

The elimination of automatic closure of the MSL drain valves on detection of high MSL radiation was not specifically discussed in NEDO-31400A. DTE proposes to eliminate this automatic closure function on the basis that the MSL drain lines ultimately discharge to the main condenser. Therefore, non-condensable gases are extracted, processed, and released to the environment through the main condenser or through the offgas system. Both of these release paths are considered in NEDO-31400A and the re-analysis of the CRDA radiological consequences performed in support of this application.

The elimination of automatic closure of the inboard and outboard reactor water sample line isolation valves associated with the reactor recirculation loop B was not specifically discussed in NEDO-31400A. The re-analysis of the CRDA radiological consequences performed in support of this application continues to assume this automatic isolation of PCIS Group 2 on high radiation in the MSLRMs. Therefore, DTE will retain this isolation in TS Table 3.3.6.1-1, but will relocate it to new Function 2.d under the heading "Primary Containment Isolation." This relocation also requires an administrative change to the Completion Time of Condition A to replace "1.f" with "2.d". Leaving the PCIS Group 2 isolation under the heading "Main Steam Isolation" as Function 1.f would potentially cause confusion as it would not isolate the main steam lines (MSIVs or MSL drain valves) but only isolate the reactor water sample line valves. The new location as Function 2.d is appropriate as some of the existing Primary Containment Isolation Functions also isolate the reactor water sample line valves (i.e., Functions 2.b and 2.c). This move does not modify the function's entries under the "Applicable Modes or Other Specified Conditions," "Required Channels per Trip System," "Surveillance Requirements," or "Allowable Value" headings in Table 3.3.6.1-1; the entries will remain the same as they were under Function 1.f. However, as part of the relocation, the "Condition Referenced from Required Action C.1" will be revised. Function 1.f in Table 3.3.6.1-1 currently references Condition D, which has required actions to isolate the associated MSL in 12 hours. Since the new Function 2.d will not isolate the MSLs, the required action associated with Condition D is no longer applicable. Instead, DTE proposes new Function 2.d in Table 3.3.6.1-1 to reference Condition F, which has a required action to isolate the affected penetration flow path in 1 hour.

Referencing Condition F will result in the appropriate action to isolate the reactor water sample line with a Completion Time that is not less limiting than currently allowed. If the required action in Condition F is not taken within the Completion Time, Condition H requires transition to Mode 3 in 12 hours and Mode 4 in 36 hours. These required Mode changes are the same as currently required by Condition D if the main steam lines are not isolated within 12 hours. Therefore, the revision to use Condition F rather than Condition D is more appropriate for the new function and the associated Completion Time is not less limiting than what is currently required.

In order to add the PCIS Group 2 isolation as a new Function 2.d, the existing Function 2.d, "Manual Initiation," is renumbered as Function 2.e. This change is an administrative change only and is made to ensure that "Manual Initiation" is the last function listed in the table, consistent with other usage throughout the TS.

Consistent with the recommendations established for the analysis of the CRDA by RG 1.183 Revision 0, the updated analysis considers potential forced release via the MVPs, SJAEs, and the GSEs. The results of the analysis of these release paths demonstrate that, in addition to the existing credited trip of the MVPs, the GSEs also require a new automatic trip to ensure the calculated radiological consequences comply with 10 CFR 50.67 limits for onsite personnel and offsite public exposures.

10 CFR 50.36(c)(2)(ii), Criterion 3, requires that a TS LCO must be established for a system, structure, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Since the reconstituted CRDA dose analysis establishes that these trips are required for mitigating the radiological consequences of this postulated accident, this submittal proposes the creation of two new TS LCOs: 3.3.7.2, "Mechanical Vacuum Pump (MVP) Trip Instrumentation," and 3.3.7.3, "Gland Seal Exhauster (GSE) Trip Instrumentation."

The proposed TS 3.3.7.2 LCO would require all four channels of Main Steam Line Radiation – High Function for the MVP trip to be operable in MODES 1 and 2 when any MVP is in service, any MSL is not isolated, and reactor core thermal power is less than or equal to 10% rated thermal power (RTP). Above 10% RTP, control rod reactivity worth is reduced such that the effects of a postulated rod drop are not sufficient to cause significant fuel damage (Figure 3-9 of Reference 7.16). In MODES 3, 4, and 5 the consequences of a CRDA are not expected to result in any fuel damage or fission product release. TS 3.3.7.2 LCO Condition A allows one channel of the Main Steam Line – High Function for the MVP trip to be inoperable for 12 hours before requiring the channel to be restored or placed in the tripped condition. Under Condition B, the loss of MVP trip capability must be restored within 1 hour. If the required Completion Times of Conditions A or B are not met, Condition C requires one of the following within 12 hours: the associated MVP(s) to be isolated, the associated MVP breaker(s) to be removed from service, the MSLs isolated, or the plant to be in MODE 3. Since the MVPs are procedurally prohibited from being operated above 5% reactor power, the allowed Completion Time of 12 hours is a reasonable time in which to reach MODE 3 or otherwise exit the LCO Applicability in an orderly

manner without challenging plant systems. Four surveillance requirements (SRs) are established:

- SR 3.3.7.2.1 CHANNEL CHECK
- SR 3.3.7.2.2 CHANNEL FUNCTIONAL TEST
- SR 3.3.7.2.3 CHANNEL CALIBRATION
- SR 3.3.7.2.4 LOGIC SYSTEM FUNCTIONAL TEST including MVP breaker actuation

The Allowable Value for the high MSL radiation MVP trip will be the 3.6 times full power N-16 background value which is consistent with the value currently assigned for the TS 3.3.1.1 and 3.3.6.1 RPS reactor scram and PCIS isolation functions that are the subject of this amendment request. The nominal trip setpoint for the MVP trip will be specified and controlled in the Fermi 2 TRM. Consistent with the current licensing basis described in UFSAR Section 15.4.9, the full power N-16 background includes consideration of the increased operating background radiation that results from operation of Hydrogen Water Chemistry with the hydrogen injection rates associated with the implementation of On-Line Noble Chemistry (OLNC).

Similarly, the proposed TS 3.3.7.3 LCO would require all four channels of Main Steam Line Radiation – High Function for the new GSE trip to be operable in MODES 1 and 2 when any GSE is in service, any MSL is not isolated, and reactor core thermal power is less than or equal to 10% RTP. Above 10% RTP, control rod reactivity worth is reduced such that the effects of a postulated rod drop are not sufficient to cause significant fuel damage (Figure 3-9 of Reference 7.16). DTE proposes that the new trip of the GSEs be automatically bypassed using the same steam and feedwater flow signals that activate and de-activate the rod block function of the Rod Worth Minimizer at the Low Power Setpoint. The GSE trip would, therefore, be automatically disabled once reactor core thermal power exceeds 10% RTP in response to sensed reactor feedwater and main steam flow rates. Conversely, the GSE trip would automatically re-enable before reducing reactor core thermal power below 10% RTP. In MODES 3, 4, and 5 the consequences of a CRDA are not expected to result in any fuel damage or fission product release. Note that a similar bypass of the MVP trip is not required for the new TS 3.3.7.2 LCO since the MVPs are procedurally prohibited from operating above 5% RTP.

TS 3.3.7.3 LCO Condition A allows one channel of the Main Steam Line – High Function for the GSE trip to be inoperable for 12 hours before requiring the channel to be restored or placed in the tripped condition. Under Condition B, the loss of GSE trip capability must be restored within 1 hour. If the required Completion Times of Conditions A or B are not met, Condition C requires one of the following within 12 hours: the associated GSE(s) to be isolated, the associated GSE breaker(s) to be removed from service, the MSLs isolated, or the plant to be in MODE 3. Since the LCO will not be applicable above 10% reactor power, the allowed Completion Time of 12 hours is a reasonable time in which to reach MODE 3 or otherwise exit the LCO Applicability in an orderly manner without challenging plant systems. Four SRs are established:

- SR 3.3.7.3.1 CHANNEL CHECK
- SR 3.3.7.3.2 CHANNEL FUNCTIONAL TEST
- SR 3.3.7.3.3 CHANNEL CALIBRATION
- SR 3.3.7.3.4 LOGIC SYSTEM FUNCTIONAL TEST including GSE breaker actuation

The Allowable Value for the high MSL radiation GSE trip will be the 3.6 times full power N-16 background value which is consistent with the value currently assigned for the TS 3.3.1.1 and 3.3.6.1 RPS reactor scram and PCIS isolation functions that are the subject of this amendment request. The nominal trip setpoint for the new GSE trip will be specified and controlled in the Fermi 2 TRM. Consistent with the current licensing basis described in UFSAR Section 15.4.9, the full power N-16 background includes consideration of the increased operating background radiation that results from operation of Hydrogen Water Chemistry with the hydrogen injection rates associated with the implementation of OLNC.

The values for the Allowable Value and nominal trip setpoint will be consistent with those assigned for the trip of the MVPs that is also required to actuate for mitigating the consequences of the design basis CRDA.

Under each of the new LCOs, when a channel is placed in an inoperable status solely for performance of required surveillances, it is proposed that entry into the associated Conditions and Required Actions will be permitted to be delayed for up to 6 hours provided the associated component trip capability is maintained. This allowance for delayed Condition entry is adopted by extension from that permitted for testing of RPS trip channels based on NEDC-30851-P-A, "Supplement 2, Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," dated March 1989 (Reference 7.39), and has been approved for MVP trip instrumentation at several other operating BWRs, such as Hope Creek (Reference 7.30). This allowance is indicated by a Note in the new LCOs that is consistent with the Note used for TS 3.3.1.1 and 3.3.6.1 where the MSLRM trip and isolation functions currently reside.

The trip of the MVPs (which also includes isolation of the line valves) is currently designed to actuate in response to a nonsafety-related offgas system 2-minute delay pipe high radiation signal as described in UFSAR Section 11.4.3.8.2.13. Under the proposed change, both the existing MVP trip and the new GSE trip will be designed to actuate in response to detection of high radiation in the MSLs. Reliance on the safety-related MSLRM signal instead of the nonsafety-related 2-minute delay pipe radiation signal is consistent with other BWR designs and is an improvement over the current licensed configuration of the MVP trip circuit in that the use of the MSLRM signals ensure the MVP and GSE trip signals are generated at the time of earliest possible detection and improves the quality of the sensing circuit. Also, consistent with other BWR designs of similar functions, while the sensing circuits are maintained as safety-related, the trip circuits for these nonsafety-related components will remain nonsafety-related with appropriate electrical separation between the safety-related and nonsafety-related instrumentation.

The MSLRM high radiation alarm function in the Main Control Room (MCR) will be retained. However, in compliance with the guidelines of NEDO-31400A, the MSLRM and offgas 2-minute delay pipe alarm setpoints will be reset to correspond to 1.5 times full power N-16 background. The new alarm setpoints will include consideration of the increased operating background radiation that results from operation of Hydrogen Water Chemistry with hydrogen injection rates associated with the implementation of OLNC consistent with the current licensing basis in UFSAR Section 15.4.9.

The subsequent sections of this enclosure provide a detailed technical evaluation in support of the proposed TS changes (Section 4.0), information supporting a finding of No Significant Hazards Consideration (Section 5.0), and information supporting an Environmental Consideration (Section 6.0). Enclosure 2 provides the existing TS pages marked up to show the proposed changes that have been described above. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides existing TS Bases pages marked up to show the proposed changes. Changes to the existing TS Bases, consistent with the technical and regulatory analyses, will be implemented under the TS Bases Control Program. Enclosure 4 is provided for information only. Enclosure 5 provides a compliance matrix comparing the CRDA re-analysis to RG 1.183.

NUREG-1433, "Standard Technical Specifications, General Electric BWR/4 Plants," (Reference 7.3) contains criteria and guidance for the improved TS for GE BWR/4 plants. The proposed changes to eliminate the reactor trip and MSIV closure associated with the MSLRMs are consistent with the guidance provided in NUREG-1433, Revision 4.

4.0 TECHNICAL EVALUATION

4.1 Design Description

The safety objective of the MSLRM system is to detect the release of fission products from a gross fuel failure and, upon indication of such failure, to initiate appropriate action to limit fuel damage and control fission product releases.

The current safety design basis for the MSLRM system is described as follows:

1. The MSLRM system is designed to give prompt indication of a release of fission products from a gross fuel failure.
2. Upon detection of a release of fission products from a gross fuel failure, the MSLRM system initiates a reactor scram and initiates action to contain the fission products released from the fuel.

Four gamma-sensitive instrumentation channels monitor the gross gamma radiation from the MSLs. The detectors are physically located near the MSLs just downstream of the outboard MSIVs. The detectors are geometrically arranged so that the system is capable of detecting significant increases in radiation level from any number of MSLs in operation. Their location along the MSLs allows the earliest practical detection of a gross fuel failure. Two of the

channels are powered from one RPS bus, and the other two channels are powered from the other RPS bus. When a significant increase in the MSL radiation level is detected, trip signals are transmitted to the RPS and PCIS. Upon receipt of the high radiation trip signals, the RPS initiates a reactor scram and the PCIS initiates closure of all MSIVs, the MSL drain valves, and the reactor water sample line isolations valves.

The setting is low enough that the MSLRMs can respond to the fission products released during the design basis CRDA. Conversely, the setting is sufficiently high enough above the rated full power N-16 background radiation level in the vicinity of the MSLs to preclude inadvertent trips due to normal variations in background radiation at rated power.

The four instrumentation channels are arranged in a one-out-of-two-twice logic to provide the required redundancy while minimizing the potential for inadvertent scram and isolation as a result of instrumentation malfunctions. The output trip signals of each monitoring channel are combined in such a way that at least two channels must signal high radiation to initiate a reactor scram and PCIS isolations. Thus, failure of any one monitoring channel does not result in inadvertent actuation.

Each monitoring channel consists of a gamma-sensitive ion chamber and a Logarithmic Radiation Monitor. Each monitor has two trip circuits: an upscale trip and an inoperable setting, either one is capable of initiating the reactor scram and PCIS isolations. The output from each monitor is displayed in the MCR.

The trip circuits for each monitoring channel operate normally energized, so that power interruptions to monitoring components result in a trip signal. The environmental capabilities of the components of each monitoring channel are selected in consideration of the locations in which the components are placed.

The number and location of the detectors meet the first safety design basis previously listed. The PCIS isolations ensure containment of radioactive materials to satisfy the second safety design basis.

The MSLRM high radiation RPS scram function is not credited to shut down the reactor in response to a postulated CRDA; instead, the neutron monitoring system is credited as the means to shut down the reactor in response to the high flux condition that results from the reactivity inserted by the CRDA.

4.2 General NEDO-31400A Approach

The BWROG issued NEDO-31400A (Reference 7.1) to provide licensees guidelines for eliminating the RPS automatic reactor trip and MSIV closure functions initiated by the MSLRMs. The NEDO-31400A approach demonstrated these functions were not required to achieve acceptable offsite radiological consequences and that given sufficiently low thresholds for radiation monitoring alarm setpoints, procedural guidance would be adequate to prompt timely operator actions to control and terminate a release due to a postulated CRDA.

In the letter dated May 15, 1991 (Reference 7.2), the NRC stated that removal of the MSLRM system high radiation trip function from initiating an automatic reactor scram and closure of the MSIVs is acceptable; however, licensees referencing NEDO-31400A in support of their TS change requests must meet the following conditions.

- Condition 1 *The applicant demonstrates that the assumptions with regard to input values (including power per assembly, Chi/Q , and decay times) that are made in the generic analysis bound those for the plant.*
- Condition 2 *The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.*
- Condition 3 *The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.*

The manner in which DTE meets these three conditions is described in the following subsections.

4.3 DTE Response to NEDO-31400A Condition 1

“The applicant demonstrates that the assumptions with regard to input values (including power per assembly, Chi/Q , and decay times) that are made in the generic analysis bound those for the plant.”

The Fermi 2 licensing basis analysis currently presented in UFSAR Section 15.4.9, “Control Rod Drop Accident,” (Reference 7.18) is based on the traditional NUREG-75/087 accident source term. Power uprates, fuel changes implemented subsequent to the issuance of NEDO-31400A as well as different assumptions regarding noble gas holdup times in the offgas system prevent the direct reliance on the results of the NEDO-31400A analysis of radiological consequences to support the proposed license changes. In addition, whereas NEDO-31400A explicitly specified evaluation of the condenser and forced SJAE release paths, current regulatory guidance (References 7.12 and 7.13) issued by the NRC after NEDO-31400A stipulates that “[if] there are forced flow paths from the turbine or condenser, such as [un-isolated] motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.” On this basis, more recent NRC-approved updated CRDA analyses, such as Peach Bottom (Reference 7.5), have

expanded the scope of forced release paths evaluated to include the MVPs, SJAEs, and main turbine GSEs, which envelopes the scope of potential release paths considered in NEDO-31400A.

In accordance with the approved NEDO-31400A protocol, the offsite and control room radiological consequences of a postulated CRDA have been re-analyzed. The radiological consequences of this accident are evaluated in accordance with the NRC staff assumptions and methods described in Appendix C of the RG 1.183 Revision 0 AST in order to conservatively demonstrate compliance with 10 CFR 50.67. Since Fermi 2 has only selectively implemented the AST methodology for analysis of the design basis fuel handling accident and LOCA (References 7.14 and 7.15), this submittal also requests approval to extend the AST methodology to the analysis of the Fermi 2 CRDA. A compliance matrix comparing this CRDA re-analysis to RG 1.183 is provided in Enclosure 5.

Forced Releases:

As previously indicated, the updated analysis considers potential forced release via the SJAEs, MVPs, and the GSEs. The results of the analysis of these release paths demonstrate that, in addition to the existing credited trip of the MVPs, a new automatic trip of the GSEs is also required to ensure the calculated radiological consequences comply with 10 CFR 50.67 limits for onsite personnel and offsite public exposures.

MVPs:

There are two MVPs which are used during startup to establish a vacuum in the condenser. The MVPs may also be used to maintain condenser vacuum following a plant shutdown/scram. The MVPs are used at low reactor powers when nuclear steam flow is insufficient to operate the SJAEs. Plant procedures prohibit operation of the MVPs above 5% reactor power. The MVPs take suction from a 20-inch manifold connected to the main condenser and discharge the non-condensable gases to the 2-minute delay pipe which provides a volume and holdup period for short-lived radionuclides prior to discharging to the environment via the reactor building exhaust stack. Currently, the MVPs automatically trip, and the line valves automatically close, on detection of high radiation in the 2-minute delay pipe. There are two 2-minute delay pipe radiation monitors. A "High-High" radiation or "Failed Downscale" signal is required from both monitors in order to cause a trip of the MVPs. The MVPs, offgas 2-minute delay pipe radiation monitors, isolation valves, and associated trip logic are nonsafety-related, non-seismic components. The scoping analysis performed in support of this submittal establishes that the source concentrations and atmospheric dispersion factors for the MVP release are significantly greater than those for the delayed release via the condenser such that continued credit for the MVP trip remains essential to ensuring the CRDA consequences satisfy the RG 1.183 dose acceptance limits for the CRDA.

It is proposed that the existing Fermi 2 MVP trips be upgraded by utilizing the safety-related MSLRM signals instead of the nonsafety-related 2-minute delay pipe signals. The use of the safety-related MSLRMs increases the redundancy and improves the reliability of the initiating trip logic up to the interface with the MVP control circuits. Downstream of the initiating logic,

the trip function logic is neither safety-related nor single failure proof. A description of the logic is included in the TS Bases markup provided in Enclosure 4. This proposed design is similar to that described in the license amendment requests for Brunswick Unit Nos. 1 and 2 and Hope Creek Generating Station. The NRC approved these requests in SERs dated May 9, 1997 for Brunswick (Reference 7.29) and March 11, 2003 for Hope Creek (Reference 7.30).

GSEs:

The GSEs draw a steam/air mixture from the turbine glands into the gland condenser where the steam is condensed. Air and other non-condensable gases are then passed through the exhausters and into the offgas system upstream of the 2-minute delay pipe. After passing through the 2-minute delay pipe, the gland seal exhaust is discharged to the environment from the reactor building stack. Condensate from the gland steam condenser flows into a drain tank which is vented to the exhauster line. The condensate is drained to the main condenser through a level control valve. Normally, one GSE is in operation with the other GSE in auto-standby.

During normal operation, the gland sealing steam flow to the gland seal condensers is estimated to be approximately 15,000 lbm/hr, or approximately 0.1% of total steam production. If the CRDA were to occur at full power operation, the impact relative to the current predicted consequences would, therefore, be small. However, the accident consequences are only credible at low power when relative rod worths are large. The steam flow required for turbine gland sealing is relatively insensitive to reactor power, such that at low power, the relative fraction of steam directed toward the GSEs is significantly larger. Thus, under the low power conditions for which the most limiting consequences of a postulated CRDA are credible, the fraction of source term released directly to the environment via gland sealing steam has the potential to be large relative to that delivered to the main condenser. Predicted doses for this path exceed the regulatory acceptance limits. On this basis, it has been determined that the GSEs should be tripped on detection of MSLRM high radiation in a manner similar to that of the MVPs described above. Such a trip of the GSEs has been implemented at the Edwin I. Hatch plant (Reference 7.36).

The fact that the absence of a trip of the GSEs results in high predicted consequences has been entered into the Fermi 2 Corrective Action Program and appropriate compensatory measures are in place to ensure consequences remain within current regulatory limits until such time as the modification to install a GSE trip proposed in this submittal has been implemented. This condition was also the subject of reports made to the NRC under 10 CFR 50.72 (Event Notification 52342) and 10 CFR 50.73 (Licensee Event Report 2016-012).

Both the existing trip of the MVPs and the new trip of the GSEs will be initiated on the Main Steam Line Radiation – High function at the same nominal trip setpoint that currently initiates the RPS reactor scram and MSIV/MSL drain valve isolations.

The proposed LCO for the GSEs (TS 3.3.7.3) would require all four channels of Main Steam Line Radiation – High function to be operable in MODES 1 and 2, with any GSE in service, any MSL not isolated, and reactor core thermal power less than or equal to 10% RTP. A spurious trip of the GSEs at power could result in a challenge to plant operation. Since the overall intent

of the proposed change is to reduce the potential for an inadvertent scram/transient at power due to spurious actuations of the MSLRM trip circuit, DTE proposes that the trip associated with the GSEs only be required to be OPERABLE in MODES 1 and 2 with reactor core thermal power less than 10% RTP. This corresponds to the applicability of TS 3.3.2.1 LCO for the Control Rod - Rod Worth Minimizer. Above 10% RTP, control rod reactivity worth is reduced such that the effects of a postulated rod drop are not sufficient to cause significant fuel damage (Figure 3-9 of Reference 7.16), and it is proposed that the new trip of the GSEs be permitted to be bypassed. This bypass will be accomplished automatically using the same steam and feedwater flow signals that are used to actuate the automatic bypass of the Rod Worth Minimizer rod block function above the Low Power Setpoint. Conversely, the GSE trip would be automatically re-enabled in response to reducing power below either of the Rod Worth Minimizer Low Power Setpoint inputs. In MODES 3, 4, and 5 the consequences of a CRDA are not expected to result in any fuel damage or fission product release. A similar bypass of the MVP trip is not required since the MVPs are procedurally prohibited from operating above 5% reactor power.

SJAEs:

The main condenser evacuation system consists of four, two-stage SJAE units, complete with intercondensers for normal plant operation and MVPs for use during startup. Typically, two of the four SJAE units are required for normal operation. When suitable steam is available, the SJAEs are put into service to remove the gases from the main condenser after vacuum has been established in the main condenser by the MVPs. Main steam is supplied as the driving medium to the two-stage SJAEs. The first stages take suction from the main condenser and exhaust the gas vapor mixture to the intercondensers. The second stages exhaust the suction gas vapor mixture from the intercondensers to the offgas system. The steam-jet intercondensers are drained back to the main condenser. The offgas system processes the exhaust of the SJAE first to recombine hydrogen and then to pass it through a series of sand and charcoal filters to remove and holdup radioactive gas and particulate. The charcoal beds are effective in removing iodine. Assuming the higher offgas flows typical of normal startup conditions, the effective holdup times for Krypton and Xenon, the primary constituents of this effluent release, are ≥ 8 hours and ≥ 4.5 days, respectively.

Consistent with guidance in RG 1.183, the evaluation of the consequences associated with the SJAEs assume this release path is active for 24 hours following the initiating event. The analysis credits the activity filtration equipment to effectively remove iodine and particulate and to holdup noble gases. Assumed noble gas holdup times are conservatively based on the higher offgas flow rates typically observed during plant startup conditions. With credit for the proposed MVP and GSE high MSL radiation trips, the combined SJAE and condenser leakage doses are demonstrated to satisfy the CRDA dose regulatory acceptance limits stipulated in RG 1.183 for compliance with 10 CFR 50.67.

Detailed Description of Dose Analysis:

The details of the re-analysis of the Fermi 2 CRDA are provided below. The results show that, with continuing credit for the trip of the MVPs as well as the new trip of the GSEs, the fission

product release involved in the CRDA and the resulting doses are relatively small. The new analysis differs from the current UFSAR Section 15.4.9 analysis with respect to the following:

- Use of AST methodology and evaluation of consequences against the total effective dose equivalent (TEDE) dose criteria of 10 CFR 50.67.
- Evaluation in accordance with NEDO-31400A.
- Revised inputs for core source term, fuel type/number of failed fuel rods, peaking factor, and release locations and corresponding relative air concentrations (compliance with RG 1.183 is discussed in Enclosure 5).

Source Term:

1. The number of failed fuel rods is assumed to be 1,200 rods for bounding case of 10x10 GE14 fuel.
2. The core isotopic inventory available for release into the reactor coolant system is based on the ORIGEN-S computer code (Reference 7.22) assuming a core bundle average exposure of 35 gigawatt-days per metric ton of uranium (GWD/MTU) for a maximum full power operation at 3,499 megawatts thermal (MWth), or 1.003 times the current licensed thermal power level of 3,486 MWth. The assumed power level credits the improved feedwater flow measurement accuracy obtained using the ultrasonic Leading Edge Flow Meter (LEFM) as approved under License Amendment No. 196 (Reference 7.20) for the Fermi 2 Measurement Uncertainty Recapture Power Uprate Project.

The source term is scaled by a radial power peaking factor of 1.7.

Note that Fermi 2 operates with maximum bundle average exposures that are less than 60 GWD/MTU; thus, no fuel exceeds the burnup limit assumption expressed in Footnote 11 of RG 1.183. Scoping evaluations using 60 GWD/MTU and 35 GWD/MTU exposed source terms demonstrated no significant differences in calculated consequences and the analysis was performed assuming a 35 GWD/MTU source term.

Table 2 provides a summary of pre-accident core activities.

3. No credit is assumed for source term decay prior to reactor startup.
4. 10% of the core inventory of noble gases and iodine, and 12% of the core inventory of alkali metals, are released from the breached fuel gap.
5. 0.77% of the breached fuel melts releasing 100% of noble gases and 50% of the iodine contained in the melted fuel fraction.
6. The analysis assumes that the speciation of radioactive iodine released from failed fuel is 95% aerosol (particulate), 4.85% elemental, and 0.15% organic, and that all other non-noble gas isotopes are released in 100% particulate form.

Transport Assumptions:

7. The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed with the reactor coolant within the pressure vessel. 100% of all noble gases, 10% of the iodines, and 1% of alkali metal nuclides are transported to the turbine/condenser. (Note that the duration of transport was conservatively modeled to occur in 1 second.)
8. Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the alkali metal nuclides are available for release to the environment.
9. The accident release duration is 24 hours.
10. As stipulated in NEDO-31400A, the re-analysis of the CRDA evaluates the radiological consequences assuming delayed release from the main condenser and a forced release from the offgas system due to the continued operation of the SJAEs.
 - a. Main Condenser: The main condenser is assumed to release the post-CRDA activity to the turbine building at a rate of 1% per day. No credit is taken for dilution or holdup within the turbine building; however, radioactive decay during holdup in the condenser is credited. For the purposes of analyzing onsite radiological consequences, the condenser activity released to the environment is assumed to exit the turbine building ventilation stack as a zero-velocity vent release.
 - b. SJAEs: The evaluation of a release via the offgas system assumes that the MSIVs do not close and that steam flow continues for approximately 24 hours before this path is isolated. The SJAEs are placed into operation once 300 psig steam is available (greater than approximately 2.5% to 3.0% RTP). The offgas system delivers non-condensable gases in the main condenser to a series of sand filters which remove particulates and a series of charcoal adsorber beds that retain iodine and holdup the noble gases to allow the natural decay process to significantly reduce activities prior to release to the environment via the reactor building exhaust stack.
 - c. As previously indicated, the scoping analysis that was prepared in support of this submittal also evaluated consequences associated with postulated releases from the MVPs and the GSEs. The results of the scoping analysis demonstrated that, in addition to the existing credited trip of the MVPs, a new automatic trip of the GSEs is required to ensure the calculated radiological consequences comply with 10 CFR 50.67 limits on onsite personnel and offsite public exposures. The re-analysis presumes acceptance of the proposal to trip and isolate the MVPs and trip the GSEs on detection of high MSL radiation.
 - d. No credit is taken for MSIV closure, nor SJAE shutdown, prior to 24 hours.

11. Elimination of the MSL drain valves isolation on MSLRM detection of high radiation was not evaluated in NEDO-31400A. During initial phases of plant startup (i.e., reactor coolant temperature less than 200°F), the MSL drain lines are isolated. Thus, these lines would not be an expected CRDA release path. During later periods of the plant startup, the MSL drain lines are opened, as needed, to maintain heatup within the TS limits. However, the MSIVs are also open during this period (i.e., the MSLs are open to the main condenser). The evaluation of the condenser release path assumes that 100% of CRDA activity is released to the main condenser in 1 second, and therefore, the transportation of the post-CRDA activity from the reactor coolant to the main condenser either via MSLs or MSL drain lines is inconsequential. As the consequences associated with the MSL drain lines are inherently included in the consequences reported for either postulated release path, the re-analysis of the CRDA does not report consequences specific to the MSL drain lines.

Atmospheric Dispersion Factors (X/Qs):

12. The relative concentrations developed for the re-analysis of the CRDA represent a change from those used in the current UFSAR analysis. The values for offsite dispersion currently in the UFSAR are based on RG 1.3 (References 7.17 and 7.18) and control room operator doses are not currently reported.

The new AST CRDA analysis assumes control room relative air concentrations calculated using the ARCON96 computer code (Reference 7.26). The atmospheric transport of main condenser leakage to the MCR is modeled as a zero-velocity vent release from the turbine building ventilation stack in the same manner previously approved for modeling of Fermi 2 design basis post-LOCA MSIV leakage (Reference 7.15). Similarly, atmospheric dispersion of SJAE discharge from the reactor building exhaust stack to the MCR is represented as a zero-velocity vent release from that location.

The relative air concentrations used for offsite dispersion to the exclusion area boundary (EAB) and low population zone (LPZ) locations are the same ground-level release values previously obtained using PAVAN and RG 1.145 guidance (References 7.24 and 7.25) that were submitted in support of the Reference 7.15 Fermi 2 AST amendment.

ARCON96 and PAVAN are both identified in RG 1.183 Rev. 0 as acceptable methods for calculating control room and offsite atmospheric dispersion coefficients for DBA analyses. The determination of relative air concentrations performed using ARCON96 and PAVAN are based on site-specific hourly meteorology data collected between January 1995 and December 1999.

The analysis assumes no credit for automatic or manual operation of the Control Room Emergency Filtration System.

Dose Consequences:

13. Dose calculations are performed using RADTRAD (Reference 7.23).
14. Dose calculations utilize the Dose Conversion Factors from Federal Guidance Reports (FGRs) Nos. 11 and 12 (References 7.27 and 7.28).
15. The offsite dose is determined as a TEDE, which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure from all radionuclides that are significant with regard to dose consequences and the released radioactivity.
16. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is demonstrated to meet the dose acceptance limits specified in RG 1.183 Rev. 0 for complying with 10 CFR 50.67(b)(2)(i).
17. TEDE determined for the most limiting receptor at the outer boundary of the LPZ is demonstrated to meet the dose acceptance limits specified in RG 1.183 Rev. 0 for complying with 10 CFR 50.67(b)(2)(ii).
18. The radioactive material releases and radiation levels used in the MCR dose analysis are determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values.
19. The MCR dose is demonstrated to comply with the limit specified in 10 CFR 50.67(b)(2)(iii).
20. The MCR operator doses from the external cloud, containment shine, and MCR filter shine doses are not analyzed for the CRDA. The MCR external cloud is negligible due to the limited degree of core damage postulated. Also, there is no release into containment, so there is no containment shine dose. MCR emergency filtration is not credited, so there is no need to consider MCR filter shine dose.

A summary of the governing CRDA input parameters and core radionuclide inventory used to generate the CRDA accident source term are provided in Tables 1 and 2, respectively, on the following pages. The calculated radiological consequences for postulated releases via the main condenser and SJAE release paths presented in Table 3 demonstrate that, with credit for tripping and isolating the MVPs and tripping the GSEs on detection of high radiation in the MSLs by the MSLRMs, expected radiological consequences are maintained within the regulatory acceptance criteria stipulated by RG 1.183 Rev. 0 for compliance with 10 CFR 50.67.

**Table 1: Parameters and Assumptions Used in Analysis
of Radiological Consequences of the Control Rod Drop Accident**

Parameter		Value						
Reactor Power		3,499 MWth						
Radial Peaking Factor		1.7						
Number of Failed Fuel Rods (Based on GE14 10x10 fuel)		1,200 rods						
Fraction Melted Fuel Rods		0.77%						
Fission Product Release Fractions:	Gap: Noble Gas	10%						
	Iodine	10%						
	Alkali Metals	12%						
Melted Fuel:	Noble Gas	100%						
	Iodine	50%						
	Alkali Metals	25%						
Transport Fractions RPV to Condenser:	Noble Gas	100%						
	Iodine	10%						
	Alkali Metals	1%						
Transport Fractions Condenser to Environment:	Noble Gas	100%						
	Iodine	10%						
	Alkali Metals	1%						
Condenser Release Rate		1% volume/day						
Charcoal Bed Holdup†:	Krypton	8 hours						
	Xenon	4.66 days						
	Iodine/Particulate	Infinite						
MCR Ventilation Parameters:		Emergency Filtration Credit						
		Ventilated Volume						
		Shine Volume						
		Normal Makeup Flow Rate						
		None						
		252,731 ft ³						
		56,960 ft ³						
		< 4,000 cfm						
Dose Conversion Factors		FGRs 11 & 12						
Dose Acceptance Limits:		MCR						
		EAB/LPZ						
		5.0 rem TEDE						
		6.3 rem TEDE						
Atmospheric Dispersion, X/Q (s/m ³):								
	EAB		LPZ		MCR			
	X/Q	BR††	X/Q	BR	X/Q Condenser	X/Q SJAЕ	MCR BR	MCR Occupancy
0-2 hr	2.09E-4	3.5E-4	4.86E-5	3.5E-4	1.17E-3	7.33E-3	3.5E-4	1.0
2-8 hr			2.17E-5	3.5E-4	9.09E-4	5.59E-3	3.5E-4	1.0
8-24 hr			1.45E-5	1.8E-4	3.41E-4	2.35E-3	3.5E-4	1.0
24-96 hr			6.02E-6	2.3E-4	2.29E-4	1.66E-3	3.5E-4	0.6
96-720 hr			1.71E-6	2.3E-4	1.73E-4	1.26E-3	3.5E-4	0.4

† Conservatively assumes holdup based on 3x normal offgas flow (120 scfm) consistent with normal startup conditions

†† BR = Breathing Rate (m³/s)

Table 2: Fermi 2 Core Isotopic Inventory

Isotope†	Inventory (Ci/MWt)	Isotope	Inventory (Ci/MWt)	Isotope	Inventory (Ci/MWt)
Kr-85	3.736E+02	Ru-106	1.558E+04	Ba-139	4.843E+04
Kr-85m	6.693E+03	Rh-105	2.624E+04	Ba-140	4.877E+04
Kr-87	1.343E+04	Sb-127	2.278E+03	La-140	5.079E+04
Kr-88	1.863E+04	Sb-129	8.507E+03	La-141	4.422E+04
Rb-86	4.767E+01	Te-127	2.244E+03	La-142	4.320E+04
Sr-89	2.609E+04	Te-127m	3.799E+02	Ce-141	4.477E+04
Sr-90	3.295E+03	Te-129	8.084E+03	Ce-143	4.142E+04
Sr-91	3.263E+04	Te-129m	1.639E+03	Ce-144	3.790E+04
Sr-92	3.463E+04	Te-131m	5.246E+03	Pr-143	4.041E+04
Y-90	3.405E+03	Te-132	3.823E+04	Nd-147	1.800E+04
Y-91	3.387E+04	I-131	2.657E+04	Np-239	5.051E+05
Y-92	3.497E+04	I-132	3.901E+04	Pu-238	8.162E+01
Y-93	2.656E+04	I-133	5.500E+04	Pu-239	1.041E+01
Zr-95	4.575E+04	I-134	6.078E+04	Pu-240	1.826E+01
Zr-97	4.322E+04	I-135	5.235E+04	Pu-241	3.847E+03
Nb-95	4.609E+04	Xe-133	5.412E+04	Am-241	4.902E+00
Mo-99	4.988E+04	Xe-135	1.451E+04	Cm-242	1.233E+03
Tc-99m	4.428E+04	Cs-134	4.793E+03	Cm-244	5.321E+01
Ru-103	4.183E+04	Cs-136	1.463E+03	Xe-138	4.680E+04
Ru-105	2.826E+04	Cs-137	4.270E+03		

†Boldface isotopes comprise the modeled chemical groups required to be considered in accordance with Regulatory Guide 1.183, Rev. 0.

**Table 3: Post-CRDA TEDE Dose (rem)
 Receptor Location**

Release Path	MCR†	EAB	LPZ
Condenser	< 0.250	< 0.030	< 0.015
SJAE††	< 2.800	< 2.770	< 0.650
Regulatory Acceptance Limits	5.0 10 CFR 50.67	6.3 25% of 10 CFR 50.67 Per RG 1.183 Rev. 0	6.3 25% of 10 CFR 50.67 Per RG 1.183 Rev. 0

† Assumes no credit for operation of Control Room Emergency Filtration

†† The SJAE doses indicated here are conservative with respect to a postulated CRDA for the following reasons. The design basis CRDA would generate release rate alarms that would result in entry into the plant emergency operating procedures (EOPs). The Radioactive Release EOP ultimately directs the operators to shut down the reactor and isolate primary sources discharging outside primary containment. Since the SJAEs operate using primary system steam, the action to manually isolate the MSIVs would effectively shut down the SJAEs. Given the magnitude of the release, it is expected that this action would occur well-within the shortest holdup time for the noble gases. Since holdup time is a function of charcoal mass and offgas flow rate, shutting down the SJAEs effectively terminates the SJAE release such that this path would not be expected to contribute to post-accident radiological consequences. Although this position was previously accepted by the NRC in AST amendments for the Edwin I. Hatch plant (Reference 7.37), the Fermi 2 CRDA analysis does not take credit for this position to limit the doses from the SJAE path.

4.4 DTE Response to NEDO-31400A Condition 2

“The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.”

Operating procedures will be reviewed and revised as necessary to ensure operator actions limit occupational doses and environmental releases upon evidence of increased levels of radioactivity in the MSLs. These changes will be completed prior to implementation of the proposed TS changes once approved by the NRC.

4.5 DTE Response to NEDO-31400A Condition 3

“The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 background dose rate at the monitor locations, and commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.”

DTE will reset the Fermi 2 MSLRM and offgas 2-minute delay pipe radiation monitor alarms to 1.5 times the normal full power N-16 background (with consideration of hydrogen addition and OLN injection) as well as amend operating procedures as necessary to ensure prompt sampling of the reactor coolant to determine possible sources of the contamination as well as to determine the need for further corrective action. The normal full power N-16 background radiation level is determined by averaging the detector outputs over a duration specified in station procedures.

These changes will be completed prior to implementation of the proposed TS changes once approved by the NRC.

The procedures will allow discretion on the part of the MCR operator to defer the action for sampling and analysis if the alarm is determined to be the expected result of a plant transient such as that which normally occurs following a down power or change in equipment alignment for which an increase in MSL radiation and no other indications of a CRDA are evident – for example with no corresponding increase in offgas radiation level. This is consistent with the approach utilized by Fitzpatrick which was accepted by the NRC (Reference 7.38).

Note that TS Section 3.7.5, “Main Condenser Offgas,” requires the release rate of activities from the main condenser be verified within limits within four hours following a 50% increase in activity.

5.0 REGULATORY ANALYSIS

The proposed changes involve eliminating the MSLRM trip functions from initiating an automatic reactor scram and automatic closure of the MSIVs based on the evaluation provided in LTR NEDO-31400A, which was approved by the NRC, as well as the additional changes previously described. In summary, the proposed changes request approval of the following:

- Elimination of the MSLRM function for initiating a reactor trip and automatic closure of the MSIVs and MSL drain valves.
- Reassignment of the MVP trip from the nonsafety-related offgas 2-minute delay pipe radiation monitor to the safety-related MSLRM. The reassigned MVP trip will be added to the TS as LCO 3.3.7.2.
- Installation of a new trip of the GSEs that will be actuated in response to high radiation detected by the MSLRMs. The GSE trip will be automatically bypassed when reactor power is above 10% RTP in response to feedwater and main steam flow rates that are consistent with the Rod Worth Minimizer at the Low Power Setpoint. The new GSE trip will be added to the TS as LCO 3.3.7.3.
- Adoption of the RG 1.183 Rev. 0 Alternate Source Term methodology and assumptions for evaluation of the radiological consequences of the Fermi 2 CRDA.

5.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.67:

The proposed TS changes are consistent with the current regulations and thus, an exemption pursuant to 10 CFR 50.12 is not required. Conformance to the current regulations will be maintained, in particular, 10 CFR 50.67, "Accident Source Term," with the elimination of the MSLRM function to initiate an automatic reactor trip and Group 1 PCIS isolation from the plant design and TS.

Footnote 1 of 10 CFR 50.67 states that, "The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible."

Assuming the MSIVs and MSL drain valves remain open following a postulated CRDA, with credit for the existing automatic trip of the MVPs and new trip of the GSEs in response to high MSL radiation, dose consequences are in conformance with 10 CFR 50.67.

10 CFR 50.36:

The MSLRM system and its associated trip and isolation functions involving initiating an automatic reactor scram and automatic closure of the MSIVs and MSL drain valves are no longer necessary to satisfy certain 10 CFR 50.36 criteria. Specifically, the dose analysis demonstrates that Criterion 3 of 10 CFR 50.36(c)(2)(ii) is no longer applicable, as both onsite and offsite dose

consequences remain within regulatory limits, assuming no automatic MSLRM reactor trip and Group 1 PCIS isolation functions.

10 CFR 50.36(c)(2)(ii), Criterion 3, requires that a TS LCO must be established for a SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Since the reconstituted CRDA dose analysis credits MVP and GSE trips to mitigate the radiological consequences of this postulated accident, it is appropriate to create two new TS LCOs: 3.3.7.2, “Mechanical Vacuum Pump (MVP) Trip Instrumentation,” and 3.3.7.3, “Gland Seal Exhauster (GSE) Trip Instrumentation.”

NUREG-1433, Revision 4:

The proposed changes to eliminate the reactor trip and MSIV closure associated with the MSLRMs are consistent with the guidance contained in the Improved Standard TS (i.e., NUREG-1433, Revision 4, “Standard Technical Specifications, General Electric BWR/4 Plants,” April 2012). The Standard TS in NUREG-1433 do not contain functions associated with the MSLRMs in TS 3.3.1.1, “Reactor Protection System (RPS) Instrumentation,” and TS 3.3.6.1, “Primary Containment Isolation Instrumentation.” The proposed changes to relocate and retain the PCIS Group 2 Isolation and add new TS for the MVP and GSE trip instrumentation is not based on NUREG-1433, but is consistent with other plants as discussed in Section 5.2 below.

5.2 Precedent

There are numerous examples of domestic BWRs that have submitted license amendment requests and received NRC approval to eliminate the MSLRM trip and isolation functions from initiating an automatic reactor trip and automatic closure of the MSIVs based on the NRC-approved report NEDO-31400A. Some of these plants include:

- Duane Arnold Energy Center – March 1992 & November 2006 (References 7.6 and 7.7)
- Hope Creek Generating Station – August 1992 (Reference 7.8)
- Cooper Nuclear Station – March 1993 (Reference 7.9)
- Quad Cities Nuclear Power Station – October 2000 (Reference 7.10)
- Vermont Yankee – September 2002 (Reference 7.11)
- Peach Bottom Atomic Power Station – July 2015 (Reference 7.5)

Similarly, there are several examples of utilities that currently or in the past have maintained or re-introduced the high MSL radiation trip of the MVPs as a TS LCO. Examples include:

- Brunswick – May 1997 (Reference 7.29)
- Hope Creek Generating Station – March 2003 (Reference 7.30)
- Quad Cities Nuclear Power Station – December 1999 (Reference 7.31)
- Pilgrim (Reference 7.32)
- James A. Fitzpatrick (Reference 7.33)

- Dresden Unit 2 –August 2001 (Reference 7.35)

Although the current revision of NUREG-1433 does not include a standard LCO template for the MSLRM trip of the MVPs, the proposed new LCO for the MVPs and GSEs generally follows the examples cited above. It is also noted that the Fitzpatrick TS (Reference 7.33) retain the PCIS Group 2 isolation and reference Condition F rather than Condition D in TS Table 3.3.6.1-1, similar to the Fermi 2 proposal.

With respect to the proposed installation of a GSE trip, the Edwin I Hatch plant is the only utility identified that has such a trip (Reference 7.36). Similar to the design of the MVPs, the Hatch plant GSEs trip and isolate on detection of MSL high radiation. Whereas DTE proposes providing an automatic bypass of the GSE trip when reactor power is above 10% RTP, the Hatch trip is active through the full range of power operation. The purpose of the proposed Fermi 2 GSE trip bypass is to minimize the potential for inadvertent actuation of the GSE trip at rated power, thereby avoiding potential challenges to continued plant operation.

DTE agrees to establish MSLRM and offgas 2-minute delay pipe high radiation alarm setpoints at 1.5 times the normal, full power N-16 background. The procedures will require prompt reactor coolant sampling upon receipt of an alarm, if the alarm is not caused by plant evolution transients known to cause transient elevated radiation readings. This is consistent with the approach utilized by Fitzpatrick which was accepted by the NRC (Reference 7.38).

5.3 No Significant Hazards Consideration

DTE has concluded that the proposed changes to the Fermi 2 TS described above do not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards, set forth in 10 CFR 50.92, "Issuance of amendment," is provided below.

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

Response: No.

The proposed changes eliminate the MSLRM trip and isolation functions from initiating an automatic reactor scram and automatic closure of the MSIVs. The justification for eliminating the MSLRM trip and MSIV isolation functions is based on the NRC-approved evaluation provided in GE LTR NEDO-31400A, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor," dated October 1992.

The MSLRM high radiation RPS scram function has never been credited to shut down the reactor in response to a postulated CRDA; instead, the neutron monitoring system will continue to be the credited means to shut down the reactor in response to the high flux condition that results from the reactivity inserted by the CRDA.

The consequences of an accident previously evaluated, have been re-evaluated consistent with RG 1.183 Rev. 0 AST (10 CFR 50.67) for the applicable DBA (i.e., the CRDA) as stipulated in NEDO-31400A. The supporting dose analyses demonstrate that, with continued credit for the automatic trip/isolation of the MVPs as well as a new proposed automatic trip of the GSEs, the consequences of the accident are within the regulatory acceptance criteria recommended in RG 1.183 Rev. 0 for compliance with 10 CFR 50.67. As a result, the consequences of any accident previously evaluated are not significantly increased.

The proposed modification of the trip logic for the MVPs to utilize the safety-related MSLRM signals is an improvement over the current licensed configuration of the MVP trip, which utilizes the nonsafety-related offgas 2-minute delay pipe radiation monitor "High-High" radiation signal. Reliance on the safety-related MSLRM signal is consistent with similar approved license amendments and, in addition to improving the quality and reliability of the sensing circuit, ensures the signal is generated at the time of earliest possible detection and therefore improves the effectiveness of the actuation. The trip setpoint utilized corresponds to the same value previously assigned for initiating MSIV isolation in response to the design basis CRDA. The offgas 2-minute delay pipe radiation monitor alarm function is being retained, with a more conservative setpoint, to continue to provide indication of increased radiation.

Similar to the MVPs, the proposed new trip of the nonsafety-related GSEs is also necessary to ensure calculated radiological consequences remain within the regulatory acceptance limits. Reliance on the safety-related MSLRM signal is consistent with BWR design for reliable tripping of the nonsafety-related MVPs and ensures the signal is reliably generated at the time of earliest possible detection and maximizes the effectiveness of the actuation.

The proposed changes also include the elimination of the MSLRM isolation function from automatically closing the MSL drain valves. The contents of the MSL drain lines are conveyed to the main condenser. The evaluation of the condenser release path assumes that 100% of CRDA activity released is transported to the main condenser in 1 second, and therefore, the transportation of the post-CRDA activity from the reactor coolant to the main condenser either via MSLs or MSL drain lines is inconsequential and is supported by the dose analyses performed in support of this submittal.

Neither the MSLRMs nor the MVPs are postulated initiators of any accident previously evaluated. None of the proposed changes alter the probability of the occurrence of the CRDA initiating event.

The loss of the GSEs is a malfunction of equipment considered in UFSAR Section 15.12 "Malfunction of Turbine Gland Sealing System." In the event that the operating blower malfunctions, the backup blower will automatically assume the gas removal requirements. Assuming loss of both blowers, vacuum will be lost in the gland steam condenser. No cladding perforations result from a malfunction of the turbine gland sealing system. The pressure in the gland steam exhaust header will increase to greater than atmospheric, allowing sealing steam to escape into the turbine building. If exhauster vacuum falls below a

specified value, caused for example by loss of alternating current (AC) power, a vacuum switch initiates the closing of the live steam supply to the gland steam header. Above 50% to 60% reactor power, the turbine is self-sealing; hence, the packing lines would remain pressurized under normal operating conditions.

The logic associated with the new trip of the GSEs will be designed to preserve the existing ability of the backup exhauster to automatically respond to a loss of the operating exhauster, in the absence of a valid high MSL radiation trip signal. Similar to the design of the RPS trip logic that is proposed to be eliminated, the GSE trip logic will be configured such that no single failure of a MSLRM can generate a GSE trip signal. As specified in the "Applicability" section for the new proposed LCO 3.3.7.3, the trip logic will be automatically bypassed when reactor power is above 10% RTP when the consequences postulated in association with a CRDA are not credible. On the basis of the configuration of the GSE trip logic, the quality of the initiating trip logic signal, and the short duration of normal operation for which the GSE trip logic will be active, the probability of a malfunction of equipment leading to the loss of the turbine gland sealing system is not significantly increased.

The proposed changes do not increase system or component pressures, temperatures, or flowrates for systems designed to prevent accidents or mitigate the consequences of an accident. Since these conditions do not change, the probability of a process-induced failure or malfunction of a SSC is not increased.

The addition of MVP and GSE SRs and LCOs to the TS enhances the reliability of these design functions by establishing administrative requirements for periodic verification of their operability.

The reliance on a lower assigned MSL high radiation alarm setpoint of 1.5 times the full power N-16 background will direct the control room operators to diagnose and act to mitigate conditions associated with fuel damage and release sooner than the current alarm condition which will reduce the potential consequences of a postulated release due to a CRDA.

On the basis of the above considerations, 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.

The proposed changes do not increase system or component pressures, temperatures, or flowrates. Since these conditions do not change, the likelihood of a process-induced failure or malfunction of a SSC not previously considered is not increased.

The reliance on the MVP trip to ensure acceptable dose consequences following a postulated CRDA is consistent with the original plant design and licensing bases. The re-assignment of the initiating input for the MVP trip logic to the MSLRM improves the quality and reliability of the credited trip initiating logic by relying on safety-related, redundant components. The quality of the nonsafety-related trip circuit itself is unchanged.

The reliance on the proposed trip of the GSEs is a function that is credited to ensure acceptable dose consequences following a postulated CRDA. The use of the safety-related redundant MSLRM signals and nonsafety-related trip circuit provides the same level of quality and reliability of the initiating trip logic and trip circuitry credited to trip the MVPs. These requirements provide the reliability necessary to ensure the assumptions of the analyzed CRDA remain valid.

Both the safety-related trip logic and the nonsafety-related trip circuits associated with the MVP and GSE trips will be designed to include qualified electrical isolation necessary to ensure the nonsafety-related trip circuitry cannot induce failures of or affect the reliability of the safety-related trip logic.

The new GSE trip will be designed to preserve the existing function for auto-start of the standby exhaustor in the event that the plant experiences a loss of the operating exhaustor, in the absence of a valid high MSL radiation trip signal. An installed automatic bypass of the GSE trip is actuated once steam flow and feedwater flow correspond to the same Low Power Setpoint used to disable the rod block function of the Rod Worth Minimizer during plant startup. This bypass will minimize the potential for the plant to experience a loss of both GSEs and potential ensuing turbine trip due to a failure of the new trip circuit. The status of the GSE trip bypass will be available to the control room operators and be required to be verified as a part of the plant general operating procedures for startup/shutdown.

Adding requirements for the MVP and GSE trip instrumentation in the TS will ensure that appropriate measures and requirements are in place such that any release of radioactive material released from a gross fuel failure will be contained in the main condenser and processed through the offgas system in the manner credited in the plant analysis of the CRDA.

The MSLRM trip and isolation functions being eliminated as described above are only applicable to the CRDA and no other event in the safety analysis. The proposed changes are consistent with the revised safety analysis assumptions for a CRDA as described in this license amendment request.

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

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

Response: No.

The proposed changes eliminating the MSLRM trip and isolation functions from initiating an automatic reactor scram and automatic closure of the MSIVs are justified based on the NRC-approved LTR NEDO-31400A and supporting dose analysis. The supporting dose analysis also supports the elimination of the MSL drain isolation function of the MSLRMs on the basis that with the valves open the source term associated with the analyzed release is directed to the main condenser the same as it would be via the MSLs themselves.

The methods of analysis and assumptions used to evaluate the consequences of the applicable impacted safety analysis (i.e. the CRDA) are consistent with the conservative regulatory requirements and guidance identified in Section 5.1 above and establish estimates of the EAB, LPZ, and MCR doses that comply with these criteria. Hence, there is reasonable assurance that Fermi 2, modified as proposed by this submittal, will continue to provide sufficient safety margins to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameters.

Adding requirements for the MVP and GSE high MSL radiation trips in the Fermi 2 TS will ensure that appropriate measures and requirements are in place to maintain the operability of these functions as such that any release of radioactive material from a gross fuel failure resulting from a CRDA will be contained in the main condenser and processed through the offgas system.

The proposed changes do not increase system or component pressures, temperatures, or flowrates for systems designed to prevent accidents or mitigate the consequences of an accident.

The analyses performed in accordance with the specified NRC-approved methods and assumptions demonstrate that the removal of the trip and isolation functions as described will not cause a significant reduction in the margin of safety, as the resulting offsite dose consequences are being maintained within regulatory limits. The proposed changes do not exceed or alter a design basis or a safety limit for a parameter to be described or established in the UFSAR.

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

5.4 Conclusion

There are no changes being proposed in this license amendment request such that commitments to applicable regulatory requirements and guidance documents described above would come into question. The evaluations documented above confirm that DTE will continue to comply with all applicable regulatory requirements. 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 NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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

6.0 ENVIRONMENTAL CONSIDERATION

These proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, in accordance with 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

- 7.1 General Electric Report NEDO-31400A, “Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor,” dated October 1992
- 7.2 Letter from A. C. Thadani (U. S. Nuclear Regulatory Commission) to G. J. Beck (BWROG), “Acceptance for Referencing Topical Report NEDO-31400,” dated May 15, 1991
- 7.3 NUREG-1433, Volume 1, Revision 4, “Standard Technical Specifications, General Electric BWR/4 Plants,” dated April 2012 (ML12104A192)
- 7.4 Institute for Nuclear Power Operations Consolidated Event System (ICES) Report #314400, “Peach Bottom Unit 3, Main Steam Line Radiation Monitor Blown Fuse Results in Half Scram and Half Group Containment Isolation Signal,” dated December 7, 2014
- 7.5 Letter from NRC to Exelon Nuclear, “Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Eliminate Main Steam Line Radiation Monitor Trip and Isolation Function,” dated July 28, 2015 (ML15167A456)
- 7.6 Letter from NRC to Iowa Electric Light and Power Company, “Amendment No. 182 to Facility Operating License No. DPR-49,” dated March 24, 1992 (ML021910064)
- 7.7 Letter from NRC to Mr. Gary Van Middlesworth, “Duane Arnold Energy Center - Issuance of Amendment Regarding Elimination of Main Steam Line Radiation Monitor Trip Function,” dated November 15, 2006 (ML063100647)

- 7.8 Letter from NRC to Public Service Electric & Gas Company, "Main Steam Line Radiation Monitor Amendment, Hope Creek Generating Station," dated August 17, 1992 (ML011760514)
- 7.9 Letter from NRC to Nebraska Public Power District, "Cooper Nuclear Station - Amendment No. 158 to Facility Operating License No. DPR-46," dated March 2, 1993 (ML021370572)
- 7.10 Letter from NRC to Commonwealth Edison Company, "Quad Cities - Issuance of Amendments," dated October 13, 2000 (ML003761321)
- 7.11 Letter from NRC to Mr. Jay K. Thayer, "Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Main Steam Line Radiation Monitor," dated September 18, 2002 (ML022480469)
- 7.12 NRC Regulatory Guide 1.183 Revision 0, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors," dated July 2000 (ML003716792)
- 7.13 NRC Regulatory Guide 1.195 Revision 0, "Methods and Assumptions For Evaluating Radiological Consequences of Design Basis Accidents At Light-Water Nuclear Power Reactors," dated May 2003 (ML031490640)
- 7.14 Letter from U. S. Nuclear Regulatory Commission, "Fermi 2 - Issuance of Amendment Re: Reevaluation of Fuel Handling Accident, Selective Implementation of 10 CFR Part 50.67 (TAC No. MB0956)," dated September 28, 2001 (ML012290521)
- 7.15 Letter from U. S. Nuclear Regulatory Commission, "Fermi 2 - Issuance of Amendment Re: Selective Implementation of Alternative Radiological Source Term Methodology (TAC No. MB7794)," dated September 28, 2004 (ML042430179)
- 7.16 NEDO-10527, "Rod Drop Accident Analysis For Large Boiling Water Reactors," General Electric, dated March 1972
- 7.17 Letter from U. S. Nuclear Regulatory Commission, "Fermi-2 - Amendment No. 87 To Facility Operating License No. NPF-43 (TAC No. M82102)," dated September 9, 1992 (ML020720520)
- 7.18 UFSAR Section 15.4.9, Control Rod Drop Accident
- 7.19 NEDC-32868P, Revision 1, "GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR-II)", dated September 2000.

- 7.20 Letter from U. S. Nuclear Regulatory Commission, "Fermi 2 - Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC No. MF0650)," dated February 10, 2014 (ML13364A131)
- 7.21 U. S. Nuclear Regulatory Commission Standard Review Plan SRP-15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, dated July 2000 (ML003734190)
- 7.22 O. W. Herman and C. V. Parks, "ORIGEN-S: SCALE System Module To Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup And Decay, And Associated Radiation Source Terms," Oak Ridge National Laboratory/NUREG/CSD-2N2/R6, Volume 2, Section F7, dated September 1998
- 7.23 S. L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, U. S. Nuclear Regulatory Commission, dated April 1998
- 7.24 T. J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, U. S. Nuclear Regulatory Commission, dated November 1982 (ML12045A149)
- 7.25 U. S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, dated November 1982 (ML003740205)
- 7.26 J. V. Ramsdell and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, U. S. Nuclear Regulatory Commission, May 1997 (ARCON)
- 7.27 Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors For Inhalation, Submersion, And Ingestion," EPA-520/1-88-020, dated September 1988
- 7.28 Federal Guidance Report No. 12, "External Exposure To Radionuclides In Air, Water, And Soil," EPA-402-R-93-081, dated September 1993
- 7.29 Letter from NRC to Carolina Power & Light Company, "Issuance Of Amendment No. 185 To Facility Operating License No. DPR-71 And Amendment No. 216 To Facility Operating License No. DPR-62 Regarding Condenser Vacuum Pump Isolation Instrumentation - Brunswick Steam Electric Plant, Units 1 And 2 (TAC Nos. M98178 And M98179)," dated May 9, 1997 (ML020360207)

- 7.30 Letter from NRC to PSEG Nuclear LLC, "Hope Creek Generating Station - Issuance Of Amendment Adding A Main Vacuum Pump Trip Specification (TAC No. MB3773)," dated March 11, 2003, (ML030560952)
- 7.31 Letter from Commonwealth Edison Company [Quad Cities] to U. S. Nuclear Regulatory Commission, "Request for an Amendment to Technical Specifications For Elimination of Main Steam Line Radiation Monitor Isolation and Scram Functions," dated December 30, 1999 (ML003671473)
- 7.32 Appendix A To Facility Operating License DPR-35 Technical Specification And Bases For Pilgrim Nuclear Power Station (ML052720275)
- 7.33 Appendix A, Technical Specifications For James A. Fitzpatrick Nuclear Power Plant Docket No. 50-333 (ML052720287)
- 7.34 Letter from Commonwealth Edison Company [Dresden Units 2 and 3] to U. S. Nuclear Regulatory Commission, "Request for Technical Specifications Change Main Steam Line Radiation Monitor Trip of the Mechanical Vacuum Pump," dated September 1, 2000 (ML003748957)
- 7.35 Letter from U. S. Nuclear Regulatory Commission to Exelon Nuclear, "Dresden Nuclear Power Station, Unit Nos. 2 And 3 – Issuance Of Amendment Re: Mechanical Vacuum Pump Trip Instrumentation (TAC Nos. MB0037 and MB0038)," dated August 16, 2001 (ML011980499)
- 7.36 Letter from U. S. Nuclear Regulatory Commission to Georgia Power Company, "Issuance of Amendments – Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC Nos. M84786 and M84787)," dated August 17, 1993 (ML012990314)
- 7.37 Letter from U. S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Regarding Alternate Source Term (TAC Nos. MD2934 and MD2935)," dated August 28, 2008 (ML081770075)
- 7.38 Letter from NRC, "Implementation of Commitments Associated with Technical Specification Amendment 207 for James A. Fitzpatrick Nuclear Power Plant (TAC No. M86978)," dated December 29, 1994 (ML010950303)
- 7.39 NEDC-30851-P-A, "Supplement 2, Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," dated March 1989

**Enclosure 2 to
NRC-17-0012**

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

**License Amendment Request to Revise Technical Specifications to
Eliminate Main Steam Line Radiation Monitor Reactor Trip and
Primary Containment Isolation System Group 1 Isolation Functions**

Marked-up Pages of Existing Fermi 2 TS

Add text:

3.3.7.2 Mechanical Vacuum Pump (MVP) Trip Instrumentation 3.3-70a
3.3.7.3 Gland Seal Exhauster (GSE) Trip Instrumentation 3.3-70d

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
<div data-bbox="61 514 198 625" style="border: 1px solid red; padding: 2px; display: inline-block;"> Replace text with: Deleted </div> H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Isolate all main steam lines. OR H.2 Be in MODE 3.	12 hours 12 hours
	I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours

(continued)

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
c. Neutron Flux - 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	≤ 120% RTP
d. Inop	1.2	3(c)	G	SR 3.3.1.1.12	NA
e. 2-out-of-4 Voter	1.2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.19	NA
f. OPRM Upscale	≥ 25% RTP	3(c)	J	SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	NA
3. Reactor Vessel Steam Dome Pressure-High	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	≤ 1113 psig
4. Reactor Vessel Water Level-Low, Level 3	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	≥ 171.9 inches
5. Main Steam Isolation Valve-Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 12% closed
6. Main Steam Line Radiation-High	1.2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 3.6 X full power background
7. Drywell Pressure-High	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	≤ 1.88 psig

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(continued)

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

Primary Containment Isolation Instrumentation
3.3.6.1

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

NOTES

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	<p>12 hours for Functions 1.f, 2.a, 2.c, 6.b, 7.a, and 7.b</p> <p style="text-align: center;"><u>AND</u></p> <p>24 hours for Functions other than Functions 1.f, 2.a, 2.c, 6.b, 7.a, and 7.b</p>
<p>-----NOTE-----</p> <p>With a Table 3.3.6.1-1 Function 5.c channel inoperable, isolation capability is considered maintained provided Function 5.b is OPERABLE in the affected room.</p> <p>-----</p> <p>B. One or more automatic Functions with isolation capability not maintained.</p>	B.1 Restore isolation capability.	1 hour

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Primary Containment Isolation Instrumentation
3.3.6.1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated main steam line (MSL).	12 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1 Isolate the affected penetration flow path(s).	24 hours

(continued)

There is no change to this page.
It is provided for information only.

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level—Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 24.8 inches
b. Main Steam Line Pressure—Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 736 psig
c. Main Steam Line Flow—High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 118.4 psid
d. Condenser Pressure—High	1, 2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 7.05 psia
e. Main Steam Tunnel Temperature—High	1,2,3	2 per trip string	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 206°F
f. Main Steam Line Radiation—High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 3.6 x full power background
g. Turbine Building Area Temperature—High	1,2,3	4	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 206°F
h. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

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(a) Except when bypassed during reactor shutdown or for reactor startup under administrative control.

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 2 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level—Low, Level 3	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 171.9 inches
b. Reactor Vessel Water Level—Low, Level 2	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 103.8 inches
c. Drywell Pressure—High	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
d. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA
d. Main Steam Line Radiation - High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 3.6 x full power background
b. HPCI Steam Supply Line Pressure—Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure—High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 20 psig
d. HPCI Equipment Room Temperature—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 162°F
e. Drywell Pressure—High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
f. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

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Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	1.2.3.(a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5 SR 3.3.7.1.6	≥ 103.8 inches
2. Drywell Pressure - High	1.2.3	2	B	SR 3.3.7.1.1 SR 3.3.7.1.3 SR 3.3.7.1.4 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 1.88 psig
3. Fuel Pool Ventilation Exhaust Radiation - High	1.2.3. (a).(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.3 SR 3.3.7.1.5 SR 3.3.7.1.6	≤ 6 mR/hr
4. Control Center Normal Makeup Air Radiation - High	1.2.3. (a).(b)	1	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.5	≤ 5 mR/hr

- (a) During operations with a potential for draining the reactor vessel.
(b) During movement of recently irradiated fuel assemblies in the secondary containment.

*There is no change to this page.
After this page, insert new Technical Specifications
3.3.7.2 and 3.3.7.3 as shown on the following pages.*

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump (MVP) Trip Instrumentation

LC0 3.3.7.2 Four channels of Main Steam Line Radiation – High Function for the MVP trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with any MVP in service, any main steam line not isolated, and THERMAL POWER ≤ 10% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of a non-functional MVP breaker. ----- Place channel in trip.	12 hours
B. MVP trip capability not maintained.	B.1 Restore trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Isolate the associated MVP(s).	12 hours
	<u>OR</u>	
	C.2 Remove the associated MVP breaker(s) from service.	12 hours
	<u>OR</u>	
	C.3 Isolate the main steam lines.	12 hours
<u>OR</u>		
C.4 Be in MODE 3.	12 hours	

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided MVP trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be $\leq 3.6 \times$ full power background.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including MVP breaker actuation.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7.3 Gland Seal Exhauster (GSE) Trip Instrumentation

LC0 3.3.7.3 Four channels of Main Steam Line Radiation – High Function for the main turbine GSE trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with any GSE in service, any main steam line not isolated, and THERMAL POWER ≤ 10% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of a non-functional GSE breaker. ----- Place channel in trip.	12 hours
B. GSE trip capability not maintained.	B.1 Restore trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Isolate the associated GSE(s).	12 hours
	<u>OR</u>	
	C.2 Remove the associated GSE breaker(s) from service.	12 hours
	<u>OR</u>	
	C.3 Isolate the main steam lines.	12 hours
	<u>OR</u>	
	C.4 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided GSE trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.7.3.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be $\leq 3.6 \times$ full power background.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including GSE breaker actuation.	In accordance with the Surveillance Frequency Control Program

**Enclosure 3 to
NRC-17-0012**

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

**License Amendment Request to Revise Technical Specifications to
Eliminate Main Steam Line Radiation Monitor Reactor Trip and
Primary Containment Isolation System Group 1 Isolation Functions**

Clean Pages of Fermi 2 TS with Changes Incorporated

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. Deleted		
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours

(continued)

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
c. Neutron Flux - 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	≤ 120% RTP
d. Inop	1,2	3 ^(c)	G	SR 3.3.1.1.12	NA
e. 2-out-of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.12 SR 3.3.1.1.17 SR 3.3.1.1.19	NA
f. OPRM Upscale	≥ 25% RTP	3 ^(c)	J	SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	NA
3. Reactor Vessel Steam Dome Pressure - High	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	≤ 1113 psig
4. Reactor Vessel Water Level - Low, Level 3	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	≥ 171.9 inches
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.9 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 12% closed
6. Deleted					
7. Drywell Pressure - High	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	≤ 1.88 psig

(continued)

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

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LC0 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
 2. Separate Condition entry is allowed for each channel.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.c, 2.d, 6.b, 7.a, and 7.b <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.c, 2.d, 6.b, 7.a, and 7.b
<p>-----NOTE----- With a Table 3.3.6.1-1 Function 5.c channel inoperable, isolation capability is considered maintained provided Function 5.b is OPERABLE in the affected room. -----</p> <p>B. One or more automatic Functions with isolation capability not maintained.</p>	B.1 Restore isolation capability.	1 hour

(continued)

Primary Containment Isolation Instrumentation
3.3.6.1

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≥ 24.8 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 736 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 118.4 psid
d. Condenser Pressure – High	1, 2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 7.05 psia
e. Main Steam Tunnel Temperature – High	1,2,3	2 per trip string	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 206°F
f. Deleted					
g. Turbine Building Area Temperature – High	1,2,3	4	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 206°F
h. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

(continued)

(a) Except when bypassed during reactor shutdown or for reactor startup under administrative control.

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 171.9 inches
b. Reactor Vessel Water Level – Low, Level 2	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 103.8 inches
c. Drywell Pressure – High	1,2,3	2	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
d. Main Steam Line Radiation – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 3.6 x full power background
e. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

(continued)

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow-High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 410 inches of water with time delay ≥ 1 second, and ≤ 5 seconds
b. HPCI Steam Supply Line Pressure-Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 90 psig
c. HPCI Turbine Exhaust Diaphragm Pressure-High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 20 psig
d. HPCI Equipment Room Temperature-High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 162°F
e. Drywell Pressure-High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
f. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

(continued)

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow- High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 95.0 inches of water with time delay ≥ 1 second and ≤ 5 seconds
b. RCIC Steam Supply Line Pressure-Low	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 53 psig
c. RCIC Turbine Exhaust Diaphragm Pressure-High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 20 psig
d. RCIC Equipment Room Temperature-High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 162°F
e. Drywell Pressure- High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
f. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

(continued)

Primary Containment Isolation Instrumentation
3.3.6.1

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 63.4 gpm
b. Area Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 183°F
c. Area Ventilation Differential Temperature – High	1,2,3	(d)	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 53°F
d. SLC System Initiation	1,2	2(b)	I	SR 3.3.6.1.5	NA
e. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 103.8 inches
f. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 95.5 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2(c)	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 171.9 inches
c. Manual Initiation	1,2,3	1 per valve	G	SR 3.3.6.1.6	NA

(continued)

- (b) SLC System Initiation only inputs into one of the two trip systems.
- (c) Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.
- (d) For Function 5.c, Reactor Water Cleanup (RWCU) System Isolation, Area Ventilation Differential Temperature - High, the required channels is 1 per room.

Primary Containment Isolation Instrumentation
3.3.6.1

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Traversing In-core Probe Isolation					
a. Reactor Vessel Water Level – Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 171.9 inches
b. Drywell Pressure – High	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig

3.3 INSTRUMENTATION

3.3.7.2 Mechanical Vacuum Pump (MVP) Trip Instrumentation

LC0 3.3.7.2 Four channels of Main Steam Line Radiation – High Function for the MVP trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with any MVP in service, any main steam line not isolated, and THERMAL POWER ≤ 10% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of a non-functional MVP breaker. ----- Place channel in trip.	12 hours
B. MVP trip capability not maintained.	B.1 Restore trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Isolate the associated MVP(s).	12 hours
	<u>OR</u>	
	C.2 Remove the associated MVP breaker(s) from service.	12 hours
	<u>OR</u>	
	C.3 Isolate the main steam lines.	12 hours
	<u>OR</u>	
	C.4 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided MVP trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.7.2.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be $\leq 3.6 \times$ full power background.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including MVP breaker actuation.	In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.7.3 Gland Seal Exhauster (GSE) Trip Instrumentation

LC0 3.3.7.3 Four channels of Main Steam Line Radiation – High Function for the main turbine GSE trip shall be OPERABLE.

APPLICABILITY: MODES 1 and 2 with any GSE in service, any main steam line not isolated, and THERMAL POWER ≤ 10% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 -----NOTE----- Not applicable if inoperable channel is the result of a non-functional GSE breaker. ----- Place channel in trip.	12 hours
B. GSE trip capability not maintained.	B.1 Restore trip capability.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Isolate the associated GSE(s).	12 hours
	<u>OR</u>	
	C.2 Remove the associated GSE breaker(s) from service.	12 hours
	<u>OR</u>	
	C.3 Isolate the main steam lines.	12 hours
<u>OR</u>		
C.4 Be in MODE 3.	12 hours	

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided GSE trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.7.3.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.2	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be $\leq 3.6 \times$ full power background.	In accordance with the Surveillance Frequency Control Program
SR 3.3.7.3.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including GSE breaker actuation.	In accordance with the Surveillance Frequency Control Program

**Enclosure 4 to
NRC-17-0012**

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

**License Amendment Request to Revise Technical Specifications to
Eliminate Main Steam Line Radiation Monitor Reactor Trip and
Primary Containment Isolation System Group 1 Isolation Functions**

Marked-up Pages of Existing Fermi 2 TS Bases (For Information Only)

Add text:

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B 3.3.7.3 Gland Seal Exhauster (GSE) Trip Instrumentation B 3.3.7.3-1

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(continued)

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as shown in the UFSAR, Figure 7.2-2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure, turbine stop valve (TSV) position, drywell pressure, ~~main steam line radiation~~, and scram discharge volume (SDV) water level as well as reactor mode switch in shutdown position and manual scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown scram signal). Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an RPS trip signal to the trip logic.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

high, a pressurization transient can occur if the MSIVs close. In MODE 2, the MSIV closure trip is automatically bypassed, and the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

Replace
text with:
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~~6. Main Steam Line High Radiation~~

~~Main Steam Line High Radiation Function ensures prompt reactor shutdown upon detection of high radiation in the vicinity of the main steam lines. High radiation in the vicinity of the main steam lines could indicate a gross fuel failure in the core. The scram is initiated to limit the fission product release from the fuel. This Function is not specifically credited in any accident analysis but is being retained for overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.~~

~~Main Steam Line High Radiation signals are initiated from four radiation monitors. Each monitor senses high gamma radiation in the vicinity of the main steam line. The Main Steam Line High Radiation Allowable Value is selected high enough above background radiation levels to avoid spurious scrams, yet low enough to promptly detect a gross release of fission products from the fuel.~~

~~Four channels of Main Steam Line High Radiation Function with two channels in each trip system, arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this function on a valid signal. This Function is required in MODES 1 and 2 where considerable energy exists such that steam is being produced at a rate which could release considerable fission products from the fuel.~~

~~The Allowable Value is based on the NRC guidelines of 3.6 times the full power background radiation level with nominal full power hydrogen injection rate. This Allowable Value remains fixed at this nominal full-power basis even when operating at reduced power and/or reduced hydrogen injection rates.~~

BASES

BACKGROUND (continued)

1. Main Steam Line Isolation

Most MSL Isolation Functions receive inputs from four channels. The outputs from these channels are combined in a one-out-of-two taken twice logic to initiate isolation of all main steam isolation valves (MSIVs). The outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate the two MSL drain valves at the containment boundary.

The exceptions to this arrangement are the Main Steam Line Flow-High Function and Area Temperature Functions. The Main Steam Line Flow-High Function uses 16 flow channels, four for each steam line. One channel from each steam line inputs to one of the four trip strings logic. Two trip strings logics make up each trip system and both trip systems must trip to cause an MSL isolation. Each trip string logic has four inputs (one per MSL), any one of which will trip the trip string logic. Either trip string logic can trip the trip system. This is effectively a one-out-of-eight taken twice logic arrangement to initiate isolation of the MSIVs. Similarly, contacts from the same four trip string logic output relays are connected into two two-out-of-two logic trip systems with each trip system isolating one of the two MSL drain valves.

The Main Steam Tunnel Temperature-High Function receives input from 16 channels. The logic is arranged similarly to the Main Steam Line Flow-High Function. The Turbine Building Area Temperature-High Function receives input from 8 channels. The inputs are arranged in a one-out-of-two trip string logic, with two trip string logics per trip system. All MSIVs will close on one-out-of-two taken twice logic from the two trip systems. Similarly, contacts from the same four trip string logic output relays are arranged in two-out-of-two logic that requires both trip systems to trip each of two MSL drain valves. Therefore, a Turbine Building Area Temperature-High condition sensed by at least one sensor in trip System A and at least one sensor in Trip System B will cause at least one of two MSL drain valves to close. At least four sensors input to four trip string logics must sense high temperature to close both MSL drain valves.

*There is no change to this page.
It is provided for information only.*

BASES

BACKGROUND (continued)

MSL Isolation Functions isolate the MSL and MSL drain isolation valves. ~~The MSL Radiation - High Function also isolates the Reactor Water Sample System.~~

Delete text

2. Primary Containment Isolation

Primary Containment Isolation Functions receive inputs from four channels. The outputs from these channels are arranged into two two-out-of-two logic trip systems. One trip system initiates isolation of all inboard primary containment isolation valves, while the other trip system initiates isolation of all outboard primary containment isolation valves. Each logic closes one of the two valves on each penetration, so that operation of either logic isolates the penetration.

Primary Containment Isolation Drywell Pressure-High and Reactor Vessel Water Level-Low, Level 3 Functions isolate lines in the drywell sumps and traversing in core probe (TIP) systems. (TIP isolation is Function 7)

Primary Containment Isolation Drywell Pressure-High and Reactor Vessel Water Level-Low Low, Level 2 Functions isolate lines in the Reactor Water Sample, Torus Water Management, Standby Gas Treatment, Combustible Gas Control, Nitrogen Inerting, and Primary Containment Monitoring Systems.

Primary Containment Isolation Drywell Pressure-High also affects isolation of lines in the RHR, CS, HPCI and RCIC systems. Primary Containment Isolation Reactor Vessel Water Level-Low Low, Level 2 Function also affects isolation of the Recirculation Pump Seal System and the Primary Containment Pneumatic Supply System.

3. and 4. High Pressure Coolant Injection System Isolation and Reactor Core Isolation Cooling System Isolation

Add text:
Primary Containment Isolation Main Steam Line Radiation - High isolates the Reactor Water Sample System.

Most Functions that isolate HPCI and RCIC receive input from two channels, with each channel in one trip system using a one-out-of-one logic. Each of the two trip systems in each isolation group is connected to one of the two valves on each associated penetration.

The exceptions are the HPCI and RCIC Turbine Exhaust Diaphragm Pressure-High, Steam Supply Line Pressure-Low,

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Each of these Functions consist of two trip strings per trip system (for a total of 4 trip strings). For the Main Steam Tunnel Temperature-High Function, each trip string has inputs from four channels. Two channels per trip string of each Function are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The ambient temperature monitoring Allowable Value is chosen to detect a feedwater line break inside the steam tunnel.

These Functions isolate the MSL and MSL drains isolation valves.

Replace text with:
Deleted

1.f. ~~Main Steam Line Radiation High~~

~~High MSL radiation indicates there is a major fission product release due to a fuel cladding failure, and could provide an active role in mitigating release due to a control rod drop accident (Ref. 2). While MSIV closure initiated by Main Steam Line Radiation High is not required to ensure compliance with those guidelines of 10 CFR 100, (Ref. 9) it is retained to maintain the overall diversity of parameters that cause an MSIV closure.~~

~~Main Steam Line Radiation High signals are initiated from steam tunnel monitors that sense the presence of excessive radiation levels, indicative of a fuel cladding failure. Four channels are available and required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.~~

~~The Allowable Value is based on the NRC guidelines of 3.6 times the full power background radiation level with nominal full power hydrogen injection rate. This allowable value remains fixed at this nominal full-power basis even when operating at reduced power and/or reduced, or eliminated, hydrogen injection rates.~~

~~This Function shares common instrumentation with the RPS.~~

~~This Function isolates the MSIVs, the MSL drains, and the Reactor Water Sample System.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure-High per Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

This Function shares common instrumentation with the RPS.

The Allowable Value was selected to be the same as the RPS Drywell Pressure-High Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates certain RHR, CS, HPCI and RCIC isolation valves, as well as groups of drywell sumps, TIP (TIP isolation is Function 7), Reactor Water Sample System, TWMS, Drywell and Suppression Pool Ventilation System, Nitrogen Inerting System, Recirculation Pump Seal System, Primary Containment Pneumatic Supply System, and PCMS isolation valves.

Add INSERT 1

2.d. Manual Initiation

Replace text with:
2.e.

The Manual Initiation channels provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the valve control.

One channel of the Manual Initiation Function per valve is available and required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the Primary Containment Isolation automatic Functions are required to be OPERABLE.

INSERT 1

2.d. Main Steam Line Radiation – High

High MSL radiation indicates there is a major fission product release due to a fuel cladding failure, and could provide indication of a control rod drop accident (Ref. 2). The isolation of the Reactor Water Sample System is required to limit doses associated with the control rod drop accident (Ref. 2).

Main Steam Line Radiation – High signals are initiated from steam tunnel monitors that sense the presence of excessive radiation levels, indicative of a fuel cladding failure. Four channels are available and required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is based on the NRC guidelines of 3.6 times the full power background radiation level with nominal full power hydrogen injection rate (Ref. 2). This allowable value remains fixed at this nominal full-power basis even when operating at reduced power and/or reduced, or eliminated, hydrogen injection rates.

This Function isolates the Reactor Water Sample System.

BASES

ACTIONS (continued)

Note 2 has been provided to modify the ACTIONS related to primary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable primary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable primary containment isolation instrumentation channel.

A.1

Delete text

Add text:
2.d,

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours for Functions 1.f, 2.a, 2.c, 6.b, 7.a, and 7.b and 24 hours for Functions other than Functions 1.f, 2.a, 2.c, 6.b, 7.a, and 7.b has been shown to be acceptable (Refs. 5 and 6) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

Delete text

Add text:
2.d,

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped

BASES

ACTIONS (continued)

channels within the same Function result in redundant automatic isolation capability being lost for the associated penetration flow path(s). The MSL Isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that both trip systems will generate a trip signal from the given Function on a valid signal. The other isolation functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two PCIVs in the associated penetration flow path can receive an isolation signal from the given Function. For Functions 1.a, 1.b, 1.d, and 1.f, this would require both trip systems to have one channel OPERABLE or in trip. For Function 1.c, this would require both trip systems to have one channel, associated with each MSL, OPERABLE or in trip. For Functions 1.e and 1.g, each Function consists of channels that monitor several locations within a given area (e.g., different locations within the main steam tunnel area). Therefore, this would require both trip systems to have one channel per location OPERABLE or in trip. For Functions 2.a, 2.b, 2.c, 3.b, 3.c, 4.b, 4.c, 5.e, and 6.b, this would require one trip system to have two channels, each OPERABLE or in trip. For Functions 3.a, 3.d, 4.a, 4.d, 5.a, 5.d, and 6.a, this would require one trip system to have one channel OPERABLE or in trip. For Functions 5.b and 5.c, each Function consists of channels that monitor several different rooms or areas. Therefore, this would require one channel per room or area to be OPERABLE (the channels are not required to be in the same trip system). As noted, with a Table 3.3.6.1-1 Function 5c channel inoperable, isolation capability is considered maintained provided Function 5.b is OPERABLE in the affected room. There is diversity in the RWCU temperature isolation instrumentation in that Area Ventilation Differential Temperature-High and the Area Temperature-High monitor for a small leak in the same rooms. The reliability of the RWCU system isolation function remains high even in the presence of single or multiple failures of differential temperature channels because a steam leak will cause a coincident trip of both the Area Ventilation Differential Temperature-High and the Area Temperature-High channels in RWCU A Pump Room, RWCU B Pump room, RWCU Phase Separator Room, and RWCU Heat Exchanger Room. There is no diversity for the RWCU Open

Replace with:
and 1.d

Add text:
2.d,

BASES

ACTIONS (continued)

Replace with:
2.e

Trench Above Pump room and RWCU Torus Room areas. The Condition does not include the Manual Initiation Functions (Functions 1.h, 2.d, 3.f, 4.f, 5.f, and 6.c), since they are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

D.1, D.2.1, and D.2.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.7

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The response time must be added to the PCIV closure times to obtain the ISOLATION SYSTEM RESPONSE TIME.

References 10 and 11 provide justification for elimination of Response Time Testing for all Primary Containment Isolation Instrumentation components except the Main Steam Line Isolation Instrumentation DC Output Relays, thus these components are required to be Response Time Tested.

The Main Steam Line Isolation Instrumentation DC Output Relays operate in parallel with the Main Steam Line Isolation Instrumentation AC Output Relays and are expected to have similar performance. The Main Steam Line Isolation Instrumentation DC Output Relays are common to Table 3.3.6.1-1, Functions 1.a, b, c, d, e, ~~f~~, and g and may be tested using any of these functions.

Delete text

ISOLATION SYSTEM RESPONSE TIME acceptance criteria for the instrumentation portion are included in Reference 7, while the acceptance criteria for the PCIV closure times are included in Reference 8. This test may be performed in one measurement, or in overlapping segments, with verification that all components are tested.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

Primary Containment Isolation Instrumentation
B 3.3.6.1

BASES

REFERENCES

1. UFSAR, Section 6.3.
2. UFSAR, Chapter 15.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.
4. UFSAR, Section 4.5.2.4.
5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
7. UFSAR, Section 7.3.
8. UFSAR, Section 6.2.
9. ~~NEDO-31400, "Safety Evaluation for Eliminating the BWR MSIV Closure Function and Scram Function of the MSL Radiation Monitor," Licensing Topical Plant Report for BWROG.~~
10. NEDO-32291, "System Analysis for Elimination of Selected Response Time Testing Requirements," January 1994; and Fermi-2 SER for Amendment 111, dated April 18, 1997.
11. NEDO-32291-A, Supplement 1, "System Analyses for The Elimination of Selected Response Time Testing Requirement," October 1999.

Replace with:
Deleted

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.7.3, "Control Room Emergency Filtration (CREF) System," overlaps this Surveillance to provide complete testing of the assumed safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Figure 9.4.2.
2. UFSAR, Section 9.4.1.
3. UFSAR, Section 6.4.1.
4. UFSAR, Chapter 15.
5. Safety Evaluation Report for Fermi Unit-2 Amendment No. 75, dated September 6, 1991.

*There is no change to this page.
After this page, insert new Technical Specification Bases
3.3.7.2 and 3.3.7.3 as shown on the following pages.*

B 3.3 INSTRUMENTATION

B 3.3.7.2 Mechanical Vacuum Pump (MVP) Trip Instrumentation

BASES

BACKGROUND

The main condenser MVP trip instrumentation initiates a trip of the MVP breakers and isolates the MVP lines following events in which radiation in the main steam lines exceeds a predetermined value. Tripping and isolating the MVPs limits the offsite and control room doses in the event of a control rod drop accident (CRDA).

The MVP trip instrumentation (Ref. 1) includes detectors, monitors, and relays that are necessary to cause initiation of a MVP trip. The channels include electronic equipment that compares measured input signals with a pre-established setpoint. When the setpoint is exceeded, the channel output relay actuates, which then outputs to the MVP trip logic.

The trip logic consists of two independent trip systems, each with two channels of Main Steam Line Radiation – High. Each trip system is a one-out-of-two logic for this Function. Thus, either channel of Main Steam Line Radiation – High in each trip system is needed to trip that system. The outputs of the channels in a trip system are then combined in a one-out-of-two taken twice logic so that both trip systems must trip to result in a MVP trip signal.

APPLICABLE
SAFETY ANALYSES

The MVP trip instrumentation is assumed in the safety analysis for the CRDA. The MVP Trip Instrumentation initiates a trip and isolation of the MVPs to limit offsite and control room doses resulting from fuel cladding failure in a CRDA (Ref. 2).

The MVP trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO The OPERABILITY of the MVP trip instrumentation is dependent on the OPERABILITY of the individual Main Steam Line Radiation – High instrumentation channels, which must have four OPERABLE channels (two in each trip system), with their setpoints within the specified Allowable Value of SR 3.3.7.2.3. Channel OPERABILITY also includes the MVP breakers.

An Allowable Value is specified for the Main Steam Line Radiation – High Function specified in the LCO. The nominal trip setpoint is specified in the Technical Requirements Manual.

APPLICABILITY The MVP trip instrumentation is required to be OPERABLE in MODES 1 and 2 when any MVP is in service (i.e. taking suction from the main condenser), any main steam line is not isolated, and THERMAL POWER \leq 10% RTP. If a postulated CRDA were to occur in MODES 1 or 2 with a MVP in service, the main steam lines not isolated, and with THERMAL POWER \leq 10% RTP, fission products released during a CRDA could be discharged directly to the environment. Therefore, the MVP trip is necessary to assure conformance with the radiological evaluation of the CRDA (Ref. 2). Above 10% RTP, the postulated effects of a CRDA are not sufficient to cause fuel damage (Ref. 2) such that the MVP trip is not required. The MVPs are also procedurally prohibited from being operated above 5% RTP to ensure the hydrogen-oxygen mixture in the MVP discharge piping remains below explosive limits. In MODES 3, 4, or 5, the consequences of a CRDA are insignificant and are not expected to result in any fuel damage or fission product releases.

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to MVP trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable MVP trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable MVP trip instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with MVP trip capability maintained (refer to Required Action B.1 Bases), the MVP trip instrumentation is capable of performing the intended function. However, the reliability and redundancy of the MVP trip instrumentation is reduced, such that a single failure in one of the remaining channels could result in the inability of the MVP trip instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive numbers of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of MVP trip, 12 hours has been shown to be acceptable (Ref. 3) to permit restoration of any inoperable channel to OPERABLE status (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As discussed in the Note, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of a non-functional MVP breaker, since placing the channel in trip may not adequately compensate for the non-functional breaker. If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in the loss of the MVPs), or if the inoperable channel is the result of a non-functional breaker, Condition C must be entered and its Required Actions taken.

BASES

ACTIONS (continued)

B.1

Condition B is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system result in not maintaining MVP trip capability. The MVP trip capability is maintained when sufficient channels are OPERABLE or in trip such that the MVP trip instrumentation will generate a trip signal from a valid Main Steam Line Radiation – High signal, and the MVP breakers will open. This would require both trip systems to have one channel OPERABLE or in trip, and the MVP breakers to be OPERABLE.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1, C.2, C.3, and C.4

If any Required Action and associated Completion Time of Condition A or B are not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours (Required Action C.4). Alternately, the associated MVP(s) may be removed from service since this performs the intended function of the instrumentation (Required Actions C.1 and C.2). An additional option is provided to isolate the main steam lines (Required Action C.3), which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser.

The allowed Completion Times of 12 hours are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

BASES

**SURVEILLANCE
REQUIREMENTS**

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided MVP trip capability is maintained. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel surveillance.

SR 3.3.7.2.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.7.2.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.2.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the MVP breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would also be inoperable.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 11.4.3.8.2.3.
2. UFSAR, Section 15.4.9.
3. NEDC-30851P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

B 3.3 INSTRUMENTATION

B 3.3.7.3 Gland Seal Exhauster (GSE) Trip Instrumentation

BASES**BACKGROUND**

The main turbine GSE trip instrumentation initiates a trip of the GSE breakers following events in which radiation in the main steam lines exceeds a predetermined value. Tripping the GSEs limits the offsite and control room doses in the event of a control rod drop accident (CRDA).

The GSE trip instrumentation (Ref. 1) includes detectors, monitors, and relays that are necessary to cause initiation of a GSE trip. The channels include electronic equipment that compares measured input signals with a pre-established setpoint. When the setpoint is exceeded, the channel output relay actuates, which then outputs to the GSE trip logic.

The trip logic consists of two independent trip systems, each with two channels of Main Steam Line Radiation – High. Each trip system is a one-out-of-two logic for this Function. Thus, either channel of Main Steam Line Radiation – High in each trip system is needed to trip that system. The outputs of the channels in a trip system are then combined in a one-out-of-two taken twice logic so that both trip systems must trip to result in a GSE trip signal.

**APPLICABLE
SAFETY ANALYSES**

The GSE trip instrumentation is assumed in the safety analysis for the CRDA. The GSE Trip Instrumentation initiates a trip of the GSEs to limit offsite and control room doses resulting from fuel cladding failure in a CRDA (Ref. 2).

The GSE trip instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO The OPERABILITY of the GSE trip instrumentation is dependent on the OPERABILITY of the individual Main Steam Line Radiation – High instrumentation channels, which must have four OPERABLE channels (two in each trip system), with their setpoints within the specified Allowable Value of SR 3.3.7.3.3. Channel OPERABILITY also includes the GSE breakers.

An Allowable Value is specified for the Main Steam Line Radiation – High Function specified in the LCO. The nominal trip setpoint is specified in the Technical Requirements Manual.

APPLICABILITY The GSE trip instrumentation is required to be OPERABLE in MODES 1 and 2 when any GSE is in service, any main steam line is not isolated, and THERMAL POWER \leq 10% RTP. If a postulated CRDA were to occur in MODES 1 or 2 with a GSE in service, the main steam lines not isolated, and with THERMAL POWER \leq 10% RTP, fission products released during a CRDA could be discharged directly to the environment. Therefore, the GSE trip is necessary to assure conformance with the radiological evaluation of the CRDA (Ref. 2). Above 10% RTP, the postulated effects of a CRDA are not sufficient to cause fuel damage (Ref. 2) such that the GSE trip is not required. In MODES 3, 4, or 5, the consequences of a CRDA are insignificant and are not expected to result in any fuel damage or fission product releases.

BASES

ACTIONS

A Note has been provided to modify the ACTIONS related to GSE trip instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable GSE trip instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable GSE trip instrumentation channel.

A.1 and A.2

With one or more channels inoperable, but with GSE trip capability maintained (refer to Required Action B.1 Bases), the GSE trip instrumentation is capable of performing the intended function. However, the reliability and redundancy of the GSE trip instrumentation is reduced, such that a single failure in one of the remaining channels could result in the inability of the GSE trip instrumentation to perform the intended function. Therefore, only a limited time is allowed to restore the inoperable channels to OPERABLE status. Because of the low probability of extensive numbers of inoperabilities affecting multiple channels, and the low probability of an event requiring the initiation of GSE trip, 12 hours has been shown to be acceptable (Ref. 3) to permit restoration of any inoperable channel to OPERABLE status (Required Action A.1). Alternately, the inoperable channel may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As discussed in the Note, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of a non-functional GSE breaker, since placing the channel in trip may not adequately compensate for the non-functional breaker. If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in the loss of the GSEs), or if the inoperable channel is the result of a non-functional breaker, Condition C must be entered and its Required Actions taken.

BASES

ACTIONS (continued)

B.1

Condition B is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system result in not maintaining GSE trip capability. The GSE trip capability is maintained when sufficient channels are OPERABLE or in trip such that the GSE trip instrumentation will generate a trip signal from a valid Main Steam Line Radiation – High signal, and the GSE breakers will open. This would require both trip systems to have one channel OPERABLE or in trip, and the GSE breakers to be OPERABLE.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1, C.2, C.3, and C.4

If any Required Action and associated Completion Time of Condition A or B are not met, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours (Required Action C.4). Alternately, the associated GSE(s) may be removed from service since this performs the intended function of the instrumentation (Required Actions C.1 and C.2). An additional option is provided to isolate the main steam lines (Required Action C.3), which may allow operation to continue. Isolating the main steam lines effectively provides an equivalent level of protection by precluding fission product transport to the condenser.

The allowed Completion Times of 12 hours are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

BASES

**SURVEILLANCE
REQUIREMENTS**

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided GSE trip capability is maintained. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 3) assumption of the average time required to perform channel surveillance.

SR 3.3.7.3.1

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.7.3.2

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

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)SR 3.3.7.3.3

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.7.3.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the GSE breakers is included as part of this Surveillance and overlaps the LOGIC SYSTEM FUNCTIONAL TEST to provide complete testing of the assumed safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel(s) would also be inoperable.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. UFSAR, Section 11.4.3.8.2.3.
2. UFSAR, Section 15.4.9.
3. NEDC-30851P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

**Enclosure 5 to
NRC-17-0012**

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

**License Amendment Request to Revise Technical Specifications to
Eliminate Main Steam Line Radiation Monitor Reactor Trip and
Primary Containment Isolation System Group 1 Isolation Functions**

Compliance Matrix: Fermi 2 CRDA Re-Analysis vs. Regulatory Guide 1.183 Rev. 0

Compliance with Regulatory Guide 1.183, Appendix C (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
Regulatory Guide 1.183, Appendix C			
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap.	Complies	10% of noble gas and 10% of core iodine are released to the reactor coolant.
	The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Complies	
2	If no or minimal ¹ fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 µCi/gm DE I-131) allowed by the technical specifications.	N/A	Fuel damage is postulated.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Complies	
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Complies	
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.	Complies	
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment.	Complies	Applicable to main condenser release path. There is no applicability to the SJA release pathway because only noble gases are released to the environment.

Compliance with Regulatory Guide 1.183, Appendix C (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
Regulatory Guide 1.183, Appendix C			
3.4 (cont.)	The turbine and condensers leak to the atmosphere as a ground-level release at a rate of 1% per day ² for a period of 24 hours, at which time the leakage is assumed to terminate.	See comments	<p>Turbine condenser leakage is assumed to occur at the Regulatory Guide 1.183 Rev. 0 Appendix C specified rate of 1% volume/day for 24 hours.</p> <p>The new AST CRDA analysis computes control room relative air concentrations using the ARCON96 computer code. The atmospheric transport of main condenser leakage to the main control room is modeled as a zero-velocity vent release from the Turbine Building HVAC stack in the same manner previously approved for modeling of Fermi 2 design basis post-LOCA MSIV leakage. (Reference 7.15 of Enclosure 1)</p> <p>The relative air concentrations used for offsite dispersion to the EAB and LPZ locations are the same ground-level release values previously obtained using PAVAN and Regulatory Guide 1.145 guidance that were submitted in support of the Reference 7.15 (of Enclosure 1) Fermi 2 AST amendment. The reactor building centerline is used as the origin.</p> <p>Due to the short duration of the postulated releases, where operation of the Control Room Emergency Filtration (CREF) is assumed, no credit is assumed for either the automatic or procedurally directed manual selection of the lower dose intake (north or south).</p>

Compliance with Regulatory Guide 1.183, Appendix C (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
Regulatory Guide 1.183, Appendix C			
3.4 (cont.)	No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.	Complies	
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	N/A	
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Complies	All iodine releases to the environment are 97% elemental and 3% organic. Differences in iodine species fractions has no bearing on dose results since both MCR and Technical Support Center (TSC) filters are equally efficient at removing any species of iodines.
Footnote 1	The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Complies	The analysis uses activity associated with projected fuel damage.

Compliance with Regulatory Guide 1.183, Appendix C (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
Regulatory Guide 1.183, Appendix C			
Footnote 2	<p>If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.</p>	See comments	<p>The analysis evaluates forced release due to the postulated continued operation of the SJAEs via the offgas system.</p> <p>Forced release via the Gland Seal Exhausters and Mechanical Vacuum Pumps were also evaluated. The results confirm the need for the existing MVP trip/isolation and establish that the GSEs will require a similar trip.</p> <p>Nominal GSE and SJAE flow rates are assumed for establishing path fractional release rates.</p> <p>With respect to SJAE offgas flow, a value of 120 scfm (3x design) is assumed as normal offgas flow rates experienced during startup are between 75-100 scfm until condenser inleakage is reduced later in the startup sequence as steam path pressures increase and turbine sealing improves. The offgas flow which is measured in units of scfm was expanded to actual CFM (ACFM) under normal condenser vacuum conditions for determining the rate of source term removal from the condenser to the offgas system. The 120 scfm representing offgas flow outside of the condenser was used to compute the reduced residence time in the offgas charcoal beds.</p>

Compliance with Regulatory Guide 1.183 (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
3.1	<p>The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.⁸ The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.⁹ The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.</p>	See comments	<p>The core isotopic inventory is based on the ORIGEN-S computer code (Reference 7.22 of Enclosure 1) assuming a core bundle average exposure of 35 GWD/MTU for a maximum full power operation at 3,499 MWth (1.003 times the licensed thermal power level of 3,486 MWth). The assumed power level credits the improved feedwater flow measurement accuracy obtained using the ultrasonic Leading Edge Flow Meter (LEFM) as approved under License Amendment No. 196 (Reference 7.20 of Enclosure 1) for the Fermi 2 Measurement Uncertainty Recapture Power Uprate Project. This source term is scaled by a radial power peaking factor of 1.7.</p> <p>Fermi 2 operates with maximum bundle average exposures < 60 GWD/MTU; thus, no fuel exceeds the burnup limit assumption expressed in Footnote 11 of RG 1.183. Scoping evaluations using 60 GWD/MTU and 35 GWD/MTU source terms demonstrated no significant differences in calculated consequences and the analysis was performed assuming a 35 GWD/MTU source term.</p> <p>Table 2 of Enclosure 1 provides a summary of pre-accident core activities.</p>

Compliance with Regulatory Guide 1.183 (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
3.1 (cont.)	For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Complies	The fission product inventory of each damaged fuel rod is determined by dividing the total core inventory by the number of effective fuel rods in the core. A radial peaking factor of 1.7 is applied to the core average inventory. This peaking factor, which is also applied in the analysis of the design basis fuel handling accident, is a constraint on the design of the Fermi 2 fuel cycle. Peaking factor is not a limit specified in the Fermi 2 Core Operating Limits Report (COLR).
	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Complies	No adjustments to the fission product inventory are made for power operations less than full power. The analysis does not address events when the facility is shut down. This represents a significant conservatism since most reactor startups typically occur after shutdown periods of a week to several weeks.
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3 [of RG 1.183]. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.	Complies	The fraction of core activity in the gap used in the radiological evaluation is from Assumption 1 of Appendix C of RG 1.183. Since Appendix C of the RG does not stipulate an acceptable value for alkali metals (i.e., Cs and Rb), the analysis assumes a gap fraction of 12% for this group based on Table 3 of RG 1.183.
3.3	For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	Complies	

Compliance with Regulatory Guide 1.183 (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
3.4	Table 5 [of RG 1.183] lists the elements in each radionuclide group that should be considered in design basis analyses.	See comments	Regulatory Guide 1.183 Rev. 0 Appendix C.1 states that assumptions regarding core inventory provided in Regulatory Position 3 are acceptable to the NRC staff. In Regulatory Position 3.4, Table 5 identifies radionuclides to be considered. A listing of radionuclides included in this analysis based on Table 5 of RG 1.183 is provided in Table 2 of Enclosure 1 of this submittal.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Complies	Iodine species are in accordance with Assumption 3.6 of Appendix C to RG 1.183. Specifically, the iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic. Iodine species factors have no impact on evaluation since the MCR and TSC filters are equally efficient at removing any iodine species.

Compliance with Regulatory Guide 1.183 (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	See comments	The mass fraction of fuel in the damaged rods is 0.0077 and is from Section 6.2.1 of NEDO-31400A.
Footnote 8	The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.	See comments	See comments related to Regulatory Position 3, 3.1, Block 1.
Footnote 9	Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.	See comments	See comments related to Regulatory Position 3, 3.1, Block 1.
Footnote 10	The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.	See comments	The reactor core source term inventory is based on a burnup of 35,000 MWD/MTU. See comments on compliance with Position 3.1 above. Fermi 2 does not utilize mixed oxide fuel. Fermi 2 operates with maximum bundle average exposures that are less than 60 GWD/MTU. The maximum bundle average exposure is 55 GWD/MTU.

Compliance with Regulatory Guide 1.183 (Assumptions for Evaluating the Radiological Consequences of a BWR Rod Drop Accident)			
RG 1.183 Section	Regulatory Guide Assumption	Fermi 2 Analysis	Comments
Footnote 11	<p>The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.</p>	See comments	<p>The analysis assumes the gap release fractions for noble gases and iodine source terms specified in Appendix C of this guide and only utilizes the Table 3 alkali metal gap fraction (12%) in the absence of specific Appendix C guidance.</p> <p>Compliance with the limitations of this footnote is evaluated preliminarily prior to the end of each operating cycle and reconfirmed at the end of the cycle.</p>